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10 CFR 50.90

June 26, 2002
2130-02-20153

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

Subject: Technical Specification Change Request No. 291 – Safety Limit Minimum Critical Power Ratio

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

In accordance with 10 CFR 50.4(b)(1), enclosed is Technical Specification Change Request No. 291.

The purpose of this Technical Specification Change Request is to revise Oyster Creek Technical Specifications to incorporate revised Safety Limit Minimum Critical Power Ratio (SLMCPR) values due to the cycle specific analysis performed by Global Nuclear Fuel for Oyster Creek Cycle 19, which will include the use of the GE11 fuel product line. The new SLMCPR values for Oyster Creek are 1.12 (three loop operation) and 1.11(for both four or five loop operation). This Technical Specification Change Request also incorporates several editorial corrections that improve readability of text without changing the meaning or intent.

Information supporting this Technical Specification Change Request is contained in Enclosure 1 to this letter, and the proposed marked up Technical Specification pages are contained in Enclosure 2. Enclosure 3 (letter from T. G. Orr (Global Nuclear Fuel) to K. Donovan (Exelon Generation Company, LLC), dated May 24, 2002) specifies the new SLMCPR values for Oyster Creek. Enclosure 3 contains information proprietary to Global Nuclear Fuel. Accordingly, it is requested that Enclosure 3 be withheld from public disclosure. An affidavit certifying the basis for this application for withholding as required by 10 CFR 2.790(b)(1) is also provided as Enclosure 5. Enclosure 4 provides a non-proprietary version of the Global Nuclear Fuel document.

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.

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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.

NRC approval of this change is requested by September 30, 2002. This requested approval date is to allow sufficient time to update the affected plant procedures and provide appropriate training prior to Cycle 19 startup.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

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

Sincerely,

06-26-02
Executed On

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid-Atlantic Regional Operating Group

Enclosures:
(1) Oyster Creek Technical Specification Change Request No. 291 Evaluation of Proposed Changes
(2) Oyster Creek Technical Specification Change Request No. 291 Markup of Proposed Technical Specification Page Changes
(3) Letter from T. G. Orr (Global Nuclear Fuel) to K. Donovan (Exelon Generation Company, LLC), dated May 24, 2002, Proprietary Version
(4) Letter from T. G. Orr (Global Nuclear Fuel) to K. Donovan (Exelon Generation Company, LLC), dated May 24, 2002, Non-Proprietary Version
(5) Global Nuclear Fuel Affidavit Certifying Request For Withholding From Public Disclosure

cc: H. J. Miller, Administrator, USNRC Region I
P. S. Tam, USNRC Senior Project Manager, Oyster Creek
R. J. Summers, USNRC Senior Resident Inspector, Oyster Creek
File No. 02047

United States of America
Nuclear Regulatory Commission

In the Matter of	)	Docket No. 50-219
AmerGen Energy Company, LLC	)	

Certificate of Service

This is to certify that a copy of Technical Specification Change Request No. 291 for the Oyster Creek Generating Station Operating License, filed with the U.S. Nuclear Regulatory Commission on June 26, 2002, has this 26th day of June 2002 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 Louis Amato
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: Michael P. Gallagher

Michael P. Gallagher
Director, Licensing & Regulatory Affairs

ENCLOSURE 1

Oyster Creek Technical Specification Change Request No. 291

Evaluation of Proposed Changes

1.0 INTRODUCTION

This letter is a request to amend Operating License No. DPR-16.

The proposed changes would revise the Operating License to: (1) incorporate the revised Safety Limit Minimum Critical Power Ratio (SLMCPR) for three loop operation and four or five loop operation due to the cycle specific analysis performed by Global Nuclear Fuel for Oyster Creek Cycle 19, and (2) incorporate several non-substantive editorial corrections to improve readability of the text. This change supports Cycle 19 operation. NRC approval of this change is requested by September 30, 2002 in order to allow sufficient time to update the affected plant procedures and provide appropriate training prior to Cycle 19 startup.

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

Revised Technical Specification Pages: 1.0-3, 1.0-7, 2.1-1, 2.1-2, 2.1-3, 2.2-1, 3.2-10, 3.5-7, 3.12-1, and 6-15.

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

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed amendment involves revising the Safety Limit Minimum Critical Power Ratio (SLMCPR) value contained in Technical Specification 2.1.A (page 2.1-1) from 1.09 to 1.12 for three recirculation loop operation and 1.11 for both four or five recirculation loop operation. The SLMCPR value is being revised for Oyster Creek based on the reload core design for Cycle 19, which will use the GE11 fuel product line. The SLMCPR values have been determined in accordance with NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Amendment 25 (GESTAR II). Amendment 25 provides the methodology for determining the cycle specific MCPR safety limits. Amendment 25 is used for determining the upcoming Cycle 19 SLMCPR values, and is intended to be used for determining future SLMCPR 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.

Technical Specification page 6-15 and Bases pages 2.1-2, 2.1-3, 2.2-1, and 3.2-10 are revised to update existing Technical Specification references to the NRC approved methodologies utilized for SLMCPR analysis. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," is currently incorporated in existing Oyster Creek Technical Specification 6.9.1.f.2.g as an approved methodology for development of core operating limits for Oyster Creek. Existing Technical Specification 6.9.1.f.2.h reference to NEDE-24195 is being deleted since this methodology will no longer be utilized for Oyster Creek.

In addition, Technical Specification pages 1.0-3, 1.0-7, 3.5-7, and 3.12-1 are being revised to correct typographical errors. These revisions are purely administrative changes that do not change the meaning or intent of the text.

3.0 BACKGROUND

The SLMCPR values have been determined in accordance with NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-14-US, June, 2000, which incorporates Amendment 25. Amendment 25 provides the methodology for determining the cycle specific MCPR safety limits. Amendment 25 is used for determining the upcoming Cycle 19 SLMCPR values. The NRC safety evaluation approving Amendment 25 is contained in a letter from the NRC to General Electric Company, 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)).

Global Nuclear Fuel has designed GE11 fuel to be in compliance with Amendment 22 incorporated in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-14-US, June 2000.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

10 CFR 50, Appendix A, General Design Criteria (GDC) 10 requires that the reactor core and associated coolant, control, and protective systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. Safety limits are required to be included in the Technical Specifications by 10 CFR 50.36. The SLMCPR is developed to assure compliance with GDC 10 for fuel cladding integrity. The SLMCPR ensures sufficient conservatism in the operating MCPR

limit that, in the event of an anticipated operational occurrence from the limiting condition of operation, at least 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. Every refueling cycle the SLMCPR is recalculated due to fuel replacement.

5.0 TECHNICAL ANALYSIS

The proposed Technical Specification change will revise Technical Specification 2.1.A to reflect the changes in the cycle specific analysis performed by Global Nuclear Fuel for Oyster Creek Cycle 19, which includes the use of the GE11 fuel product line.

The new SLMCPR values are calculated using NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U.S. Supplement, NEDE-24011-P-A-14-US, June 2000 which incorporates Amendment 25. Amendment 25 is used for determining the upcoming Cycle 19 SLMCPR values. Future SLMCPR values determined in accordance with Amendment 25 will not need prior NRC approval for each cycle unless a value changes. The NRC safety evaluation approving Amendment 25 is contained in a letter from the NRC to General Electric Company dated March 11, 1999.

Global Nuclear Fuel has designed GE11 fuel to be in compliance with Amendment 22 to “General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A-14 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-14-US, June, 2000.

The SLMCPR analysis establishes SLMCPR values that will ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The SLMCPR values are calculated to include cycle specific parameters, which include: 1) the actual core loading, 2) conservative variations of projected control blade patterns, 3) the actual bundle parameters (e.g., local peaking), and 4) the full cycle exposure range. The new SLMCPR values for Oyster Creek, Cycle 19 are 1.12 (three loop operation) and 1.11 (for both four or five loop operation) as shown in Enclosure 2. Additional information regarding the 1.12 and 1.11 cycle specific SLMCPR values for Oyster Creek Cycle 19 is contained in the Enclosure 3 Global Nuclear Fuel letter.

The combined results of these evaluations demonstrate that the proposed change is acceptable since no fuel thermal limits or other licensing basis acceptance criteria are adversely affected. The proposed change will be implemented at the beginning of operating Cycle 19.

The previously licensed methodology contained in NEDE-24195, “General Electric Reload Fuel Application for Oyster Creek,” listed in existing Technical Specification 6.9.1.f.2.h, will no longer be utilized for Oyster Creek and is therefore being deleted.

The proposed changes to update the existing Technical Specification Bases section references to the NRC approved methodologies utilized for Oyster Creek SLMCPR analysis and the proposed editorial changes to the Technical Specifications are considered administrative only and do not adversely affect nuclear safety or safe plant operations.

Conclusion

The proposed changes to implement revised SLMCPR values for Oyster Creek provide safety limit protection in compliance with GDC 10 by ensuring that 99.9% of the fuel rods in the core will not experience boiling transition, which satisfies the requirements of 10 CFR 50 Appendix A, GDC-10 regarding acceptable fuel limits. The proposed safety limit values have been developed by Global Nuclear Fuel using plant and cycle specific fuel and core parameters in accordance with NRC approved methodologies applicable to Oyster Creek. Consequently, the proposed Technical Specification changes will not adversely affect nuclear safety or safe plant operations.

6.0 REGULATORY ANALYSIS

10 CFR 50.36(c)(1) requires that safety limits be included in the plant Technical Specifications. Therefore, the SLMCPR is included in the Oyster Creek Technical Specifications. The SLMCPR values have been determined in accordance with NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-14-US, June, 2000.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

AmerGen has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

The derivation of the cycle specific Safety Limit Minimum Critical Power Ratio (SLMCPR) values for incorporation into the Technical Specifications (TS), and their 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-14 (GESTAR-II), and U.S. Supplement, NEDE-24011-P-A-14-US, June, 2000, which incorporates Amendment 25. Amendment 25 was approved by the NRC in a safety evaluation report dated March 11, 1999.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPR values preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The GE11 fuel is in compliance with Amendment 22 to "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-14-US, June, 2000, which provides the fuel licensing acceptance criteria. The proposed safety limit values have been developed by Global Nuclear Fuel using plant and cycle specific fuel and core parameters in accordance with NRC approved methodologies applicable to Oyster Creek. Neither the probability nor the consequences of fuel damage will be increased as a result of this change.

The proposed changes to the Technical Specification Bases and editorial corrections are considered administrative only and have no affect on nuclear safety or safe plant operations.

Therefore, the proposed change does 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 SLMCPR is a Technical Specification numerical value, designed to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. The new SLMCPR values are calculated using NRC approved methodology discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U.S. Supplement, NEDE-24011-P-A-14-US, June, 2000, which incorporates Amendment 25. Additionally, the GE11 fuel is in compliance with Amendment 22 to "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-14-US, June, 2000, which provides the fuel licensing acceptance criteria. The SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Technical Specification Bases and editorial corrections are considered administrative only and have no affect on nuclear safety or safe plant operations.

Therefore, the proposed change does 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.

There is no significant reduction in the margin of safety previously approved by the NRC as a result of the proposed change to the SLMCPR values, which includes the use of GE11 fuel. The new SLMCPR values are calculated using methodology discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), and U.S. Supplement, NEDE-24011-P-A-14-US, June, 2000, which incorporates Amendment 25. The SLMCPR values ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. The margin of safety, as defined in the Technical Specifications, for all events is maintained.

The proposed changes to the Technical Specification Bases and editorial corrections are considered administrative only and have no affect on nuclear safety or safe plant operations.

Therefore, the proposed change does 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(c), 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 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 effluent that may be released offsite, or (iii) a 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(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

The proposed SLMCPR changes for Oyster Creek are similar to the SLMCPR changes approved by the NRC for the Limerick Generating Station, Unit 1 in Amendment No. 156, dated March 12, 2002, and the Peach Bottom Atomic Power Station, Unit 3 in Amendment No. 233, dated October 5, 1999, with the exception that Oyster Creek is only introducing the use of GE11 fuel for Cycle 19. The Oyster Creek Cycle 19, SLMCPR analysis was performed by Global Nuclear Fuel using plant and cycle specific fuel and core parameters, and NRC approved methodologies including NEDC-32505P, Revision 1, *R-Factor Calculation Method for GE11 Fuel,* NEDO-10958-A, *General Electric BWR Thermal Analysis Basis* (GETAB), NEDC-32694P, *Power Distribution Uncertainties for Safety Limit MCPR Evaluation*, and Amendment 25 to NEDE-24011-P-A (GESTAR II).

10.0 REFERENCES

a) NEDE-24011-P-A, "General Electric Standard Application for Reload Fuel," Amendment 25 (GESTAR-II).

b) NRC Safety Evaluation Report, 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).

c) NEDE-24011-P-A-14 (GESTAR II), and U.S. Supplement, NEDE-24011-P-A-14-US, June 2000.

d) Letter from T. G. Orr (Global Nuclear Fuel) to K. Donovan (Exelon Generation Company, LLC), dated May 24, 2002 (Proprietary).

ENCLOSURE 2

Oyster Creek Technical Specification Change Request No. 291

Markup of Proposed Technical Specification Page Changes

<u>Revised TS Pages</u>

1.0-3
1.0-7
2.1-1
2.1-2
2.1-3
2.2-1
3.2-10
3.5-7
3.12-1
6-15

1.14 SECONDARY CONTAINMENT INTEGRITY

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

A. At least one door at each access opening is closed.
(Note: Momentary opening and closing of the trunnion room door does not constitute a loss of secondary containment integrity.)

B. The standby gas treatment system is operable.

C. All automatic secondary containment isolation valves are operable or are secured in the closed position.

1.15 (DELETED)

1.16 RATED FLUX

Rated flux is the neutron flux that corresponds to a steady state power level of 1930 M ~~N~~W(t). Use of the term 100 percent also refers to the 1930 thermal megawatt power level.

1.17 REACTOR THERMAL POWER-TO-WATER

Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS

A. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

B. Trip System

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).

parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4; and (2) descriptions of the information that should be included in the Annual Radioactive Effluent Release Report AND Annual Radiological Environmental Operating Report required by Specifications 6.9.1.d and 6.9.1.e, respectively.

1.37 PURGE

PURGE OR PURGING is the controlled process of discharging air or gas from a confinement and replacing it with air or gas.

1.38 SITE BOUNDARY

The SITE BOUNDARY is the perimeter line around the OCNGS beyond which the land is neither owned, leased nor otherwise subject to control by AmerGen Energy Company, LLC (ref. ODCM). The area outside the SITE BOUNDARY is termed OFFSITE or UNRESTRICTED AREA.

1.39 REACTOR VESSEL PRESSURE TESTING

System pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely water solid, core not critical and section 3.2.A satisfied.

1.40 SUBSTANTIVE CHANGES

(3)

SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Example of non-substantive changes are: (1) correcting spelling, (2) adding (but not deleting) sign-off spaces, ~~(3)~~ blocking in notes, cautions, etc, (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications, and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.

1.41 DOSE EQUIVALENT I-131

DOSE EQUIVALENT I- 131 shall be that concentration of I-131 microcuries per gram which alone would produce the same thyroid dose as the quantity and isotopic mixture of I131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 or Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluences for the Purpose of Evaluating Compliance with 10 CFR Par 40 Appendix I."

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~~ 1.11 for both four or five loop operation and 1.12 for three loop operation shall constitute violation of the fuel cladding integrity safety limit.

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.

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 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 (2) Limit MCPR (1) is determined using the General Electric Thermal Analysis Basis, GETAB, 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, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR)

(2) General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO-10958-A, January 1977.

2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

Applicability: Applies to the limit on reactor coolant system pressure.

Objective: Preserve the integrity of the reactor coolant system.

Specification: The reactor coolant system pressure shall not exceed 1375 psig whenever irradiated fuel is in the reactor vessel.

Bases:

The reactor coolant system(1) represents an important barrier in the prevention of the uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1375 psig was derived from the design pressures of the reactor pressure vessel, coolant piping, and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1200 psig at 570°F and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section I for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III for the isolation condenser and the ASA Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over the design pressure (110% x 1250 = 1375 psig) and the ASA Code permits pressure transients up to 15% over the design pressure (115% x 1200 = 1380 psig).

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 20,000 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 2 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 22,000 psi, still almost a factor of 2 below the yield strength. The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

The normal operating pressure of the reactor coolant system is 1020 psig. An overpressurization analysis (2) is peformed each cycle to assure that the pressure safety limit is not exceeded. The reactor fuel cladding can withstand pressures up to the safety limit, 1375 psig, without collapsing.(3) Finally, reactor system pressure is continuously monitored in the control room during reactor operation.

REFERENCES

(1) FDSAR, Volume I, Section IV.
(2) ~~NEDO-24195, General Electric Reload Fuel Application for Oyster Creek.~~ NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR).
(3) FDSAR, Volume I, Section III-2.3.3

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution will be maintained at least 5°F above the saturation temperature to guard against precipitation. The 5°F margin is included in Figure 3.2-2. Temperature and liquid level alarms for the system are annunciated in the control room.

The acceptable time out of service for a standby liquid control system pumping circuit as well as other safety features is determined to be 10 days. However, the allotted time out of service for a standby liquid control system pumping circuit is conservatively set at 7 days in the specification. Systems are designed with redundancy to increase their availability and to provide backup if one of the components is temporarily out of service.

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory with expected inventory based on appropriately corrected past data. Experience at Oyster Creek and other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References:

(1) FDSAR, Volume I, Section III-5.3.1
(2) FDSAR, Volume I, Section VI-3
(3) FDSAR, Volume I, Section III-5.2.1
(4) FDSAR, Volume I, Section VII-9
(5) ~~NEDO-24195, General Electric Reload Fuel Application for Oyster Creek~~ NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR).
(6) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
(7) FDSAR, Volume I, Sections VII-4.2.2 and VII-4.3.1
(8) FDSAR, Volume I, Section VI-4
(9) FDSAR, Amendment No. 55, Section 2
(10) C. J. Paone, Banked Position Withdrawal Sequence, January 1988 (NEDO-21231)
(11) UFSAR, Volume 4, Section 4.3.2.4.1

6. With one standby gas treatment system circuit inoperable:

   a. During Power Operation:

      (1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

      (2) Continue to verify the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.

      (3) Restore the inoperable standby gas treatment circuit to operable status within 7 days.

   b. During Refueling:

      (1) Verify the operability of the other standby gas treatment system within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

      (2) Continue to verify the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.

      (3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes.

7. If Specifications 3.5.B.5 and 3.5.B.6 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition of Specification 3.5.B.1 shall be met.

3.12 Alternate Shutdown Monitoring Instrumentation

<u>Applicability:</u> Applies to the operating status of alternate shutdown monitoring instrumentation.

<u>Objective:</u> To assure the operability of the alternate shutdown monitoring instrumentation.

<u>Specification:</u>

A. The alternate shutdown monitoring instruments listed in Table 3.12-1 shall be operable during reactor power operations and when reactor coolant temperature exceeds 212°F.

B. With less than the minimum number of operable channels specified in Table 3.12-1, either restore the inoperable channel to operable status within 30 days, or be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

<u>Basis:</u>

The operability of the alternate shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of hot shutdown of the plant from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with Appendix R and General Design Criteria 19 of 10 CFR 50.

c. GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

d. GPUN TR 040, Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

e. GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

f. NEDE-31462P and NEDE-31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

g. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (GESTAR II) (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

h. DELETED ~~NEDE-24195, General Electric Reload Fuel Application for Oyster Creek, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~

i. XN-75-55-(A); XN-75-55, Supplement 1-(A); XN-75-55, Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant," April 1977

j. XN-75-36(NP)-(A); XN-75-36(NP), Supplement 1-(A), "Spray Cooling Heat Transfer Phase Test Results, ENC - 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Basis: 6.9.1.e - RELOCATED TO THE ODCM

6.9.2 <u>REPORTABLE EVENTS</u>

The submittal of Licensee Event Reports shall be accomplished in accordance with the requirements set forth in 10 CFR 50.73.

ENCLOSURE 4

Letter from T. G. Orr (Global Nuclear Fuel) to K. Donovan

(Exelon Generation Company, LLC), dated May 24, 2002

Non-Proprietary Version

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

Tammy G. Orr
Exelon Account Leader

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May 24, 2002
TGO:02-008

Kevin Donovan
Nuclear Fuel Services
Exelon Nuclear

REFERENCE: "Additional Information Regarding the Oyster Creek Cycle 19 Cycle Specific SLMCPR", prepared by Hongbin Zhang and verified by Ed Gibbs, dated May 24, 2002.

SUBJECT: **Oyster Creek Cycle 19 Safety Limit MCPR**

Dear Kevin:

GNF proposes that the Oyster Creek Cycle 19 SLMCPR use the bounding values of 1.12 for three loop operation and 1.11 for both four or five loop operation as provided in the referenced attachment. These results are based on Monte Carlo calculations done for approximately every 4 GWD/MT.

Enclosed for your information and use is the referenced additional information regarding the Oyster Creek Cycle 19 cycle specific SLMCPR. Please note that the referenced attachment contains Global Nuclear Fuel Proprietary Information contained within the double brackets and should be handled in accordance with the proprietary information provisions contained in the Fuel Contract.

In addition, a non-proprietary version is also included.

If you have any questions regarding this information, please contact myself or Hongbin Zhang at (910) 675-6650.

Very truly yours,

Tammy G. Orr
Exelon Account Leader

References

[1] Letter, Frank Akstulewicz (NRC) to Glen 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), March 11, 1999.

[2] Letter, Thomas H. Essig (NRC) to Glen A. Watford (GE), "Acceptance for Referencing of Licensing Topical Report NEDC-32505P, Revision 1, *R-Factor Calculation Method for GE11, GE12 and GE13 Fuel*," (TAC No. M99070 and M95081), January 11, 1999.

[3] *General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application*, NEDO-10958-A, January 1977.

[4] Letter, G.A. Watford (GNF) to J.E. Donoghue (NRC), Final Presentation Material for GEXL Presentation - February 11, 2002; FLN-2002-004; February 12, 2002.

Comparison of Oyster Creek SLMCPR Values for Cycles 19 and 18

Table 1 summarizes the relevant input parameters and results of the SLMCPR determination for the Oyster Creek Cycle 19 and 18 cores. The SLMCPR evaluations were performed using NRC approved methods and uncertainties[1]. These evaluations yield different calculated SLMCPR values because different inputs were used. The quantities that have been shown to have some impact on the determination of the safety limit MCPR (SLMCPR) are provided.

In comparing the Oyster Creek Cycle 19 and Cycle 18 SLMCPR values it is important to note the impact of the differences in the core and bundle designs. These differences are summarized in Table 1.

In general, the calculated safety limit is dominated by two key parameters: (1) flatness of the core bundle-by-bundle MCPR distributions and (2) flatness of the bundle pin-by-pin power/R-factor distributions. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR.

[[ ]]

The uncontrolled bundle pin-by-pin power distributions were compared between the Oyster Creek Cycle 19 bundles and the Cycle 18 bundles. Pin-by-pin power distributions are characterized in terms of R-factors using the NRC approved methodology[2]. For the Oyster Creek Cycle 19 limiting case analyzed at EOC, [[ ]] the Oyster Creek Cycle 18 bundles are flatter than the bundles used for the Cycle 19 SLMCPR analysis.

Summary

[[ ]] have been used to compare quantities that impact the calculated SLMCPR value. Based on these comparisons, the conclusion is reached that the Oyster Creek Cycle 19 core has a flatter core MCPR distribution [[ ]] than what was used to perform the Cycle 18 SLMCPR evaluation; and the Oyster Creek Cycle 18 core has a flatter in-bundle power distributions [[ ]] than what was used to perform the Cycle 19 SLMCPR evaluation.

The calculated 1.11 Monte Carlo SLMCPR for Oyster Creek Cycle 19 is consistent with what one would expect [[ ]] the 1.11 SLMCPR value is appropriate.

Based on all of the facts, observations and arguments presented above, it is concluded that the calculated SLMCPR value of 1.11 for the Oyster Creek Cycle 19 core is appropriate. It is reasonable that this value is 0.03 higher than the 1.08 value calculated for the previous cycle. This value applies to both four and five loop operation.

For three loop operations (3LO) the calculated safety limit MCPR for the limiting case is 1.12 as determined by specific calculations for Oyster Creek Cycle 19.

Supporting Information

The following information is provided in response to NRC questions on similar submittals regarding changes in Technical Specification values of SLMCPR. NRC questions pertaining to how GE11 applications satisfy the conditions of the NRC SER[1] have been addressed in Reference 4. Only those items that require a plant/cycle specific response are presented below since all the others are contained in the references that have already been provided to the NRC.

The core loading information for Oyster Creek Cycles 18 and 19 is provided in Figures 1 and 2, respectively. The impact of the fuel loading pattern differences on the calculated SLMCPR is correlated to the values of [[ ]]

Table 1
Comparison of the Oyster Creek Cycle 19 and Cycle 18 SLMCPR

QUANTITY, DESCRIPTION	Oyster Creek Cycle 18	Oyster Creek Cycle 19
Number of Bundles in Core	560	560
Limiting Cycle Exposure Point	BOC / EOC	EOC
Cycle Exposure at Limiting Point [MWd/STU]	N/A	10400
Reload Fuel Type	GE9B	GE11
Latest Reload Batch Fraction [%]	32.8 %	33.9 %
Latest Reload Average Batch Weight % Enrichment	3.45 %	3.70 %
Batch Fraction for GE9	100%	66.1%
Batch Fraction for GE11	0%	33.9%
Core Average Weight % Enrichment	3.46 %	3.54 %
Core MCPR (for limiting rod pattern)	N/A	1.56
[[		]]
[[		]]
Power distribution uncertainty	GETAB	GETAB
Non-power distribution uncertainty	Revised	Revised
Calculated Safety Limit MCPR	**1.08**	**1.11**

Prepared by:

H Zhang

H. Zhang
Technical Program Manager

Verified by:

E.W. Gibbs

E.W. Gibbs
Technical Program Manager

[[ ]]
[[ ]]

Figure 1 Reference Core Loading Pattern – Cycle 18

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	
1									F	F	F	F	F	F	F	F	F	F									52
2								F	A	D	A	D	A	A	D	A	D	A	F								50
3					F	F	F	A	A	F	A	A	C	C	A	A	F	A	A	F	F	F					48
4				E	F	A	D	D	D	D	C	C	A	A	C	C	D	D	D	D	A	F	E				46
5			F	F	F	G	C	B	G	G	C	A	F	F	A	C	G	G	B	C	G	F	F	F			44
6			F	A	G	F	B	C	G	G	B	C	F	F	C	B	G	G	C	B	F	G	A	F			42
7			F	D	C	B	C	A	C	B	C	A	C	C	A	C	B	C	A	C	B	C	D	F			40
8		F	A	D	B	C	A	C	A	C	A	C	A	A	C	A	C	A	C	A	C	B	D	A	F		38
9	F	A	A	D	G	G	C	A	F	F	C	B	F	F	B	C	F	F	A	C	G	G	D	A	A	F	36
10	F	D	F	D	G	G	B	C	F	F	A	C	F	F	C	A	F	F	C	B	G	G	D	F	D	F	34
11	F	A	A	C	C	B	C	A	C	A	C	A	C	C	A	C	A	C	A	C	B	C	C	A	A	F	32
12	F	D	A	C	A	C	A	C	B	C	A	C	A	A	C	A	C	B	C	A	C	A	C	A	D	F	30
13	F	A	C	A	F	F	C	A	F	F	C	A	F	F	A	C	F	F	A	C	F	F	A	C	A	F	28
14	F	A	C	A	F	F	C	A	F	F	C	A	F	F	A	C	F	F	A	C	F	F	A	C	A	F	26
15	F	D	A	C	A	C	A	C	B	C	A	C	A	A	C	A	C	B	C	A	C	A	C	A	D	F	24
16	F	A	A	C	C	B	C	A	C	A	C	A	C	C	A	C	A	C	A	C	B	C	C	A	A	F	22
17	F	D	F	D	G	G	B	C	F	F	A	C	F	F	C	A	F	F	C	B	G	G	D	F	D	F	20
18	F	A	A	D	G	G	C	A	F	F	C	B	F	F	B	C	F	F	A	C	G	G	D	A	A	F	18
19		F	A	D	B	C	A	C	A	C	A	C	A	A	C	A	C	A	C	A	C	B	D	A	F		16
20			F	D	C	B	C	A	C	B	C	A	C	C	A	C	B	C	A	C	B	C	D	F			14
21			F	A	G	F	B	C	G	G	B	C	F	F	C	B	G	G	C	B	F	G	A	F			12
22			F	F	F	G	C	B	G	G	C	A	F	F	A	C	G	G	B	C	G	F	F	F			10
23				E	F	A	D	D	D	D	C	C	A	A	C	C	D	D	D	D	A	F	E				8
24					F	F	F	A	A	F	A	A	C	C	A	A	F	A	A	F	F	F					6
25								F	A	D	A	D	A	A	D	A	D	A	F								4
26									F	F	F	F	F	F	F	F	F	F									2
	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	

		Number in Core	Cycle Loaded
E	GE9B-P8DWB348-12GZ-80U-145-T6	4	15
F	GE9B-P8DWB348-12GZ-80U-145-T6	148	16
G	GE9B-P8DWB338-11GZ-80U-145-T6	40	16
A	GE9B-P8DWB348-12GZ-80U-145-T6	144	17
B	GE9B-P8DWB338-11GZ-80U-145-T6	40	17
C	GE9B-P8DWB348-12GZ-80U-145-T6	136	18
D	GE9B-P8DWB338-11GZ-80U-145-T6	48	18
	Total	560	

[[ ]]
[[ ]]

Figure 2 Reference Core Loading Pattern – Cycle 19

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	
1									A	A	A	A	A	A	A	A	A	A									52
2								A	C	D	C	D	D	D	D	C	D	C	A								50
3					A	A	A	A	C	C	C	C	C	C	C	C	C	C	A	A	A	A					48
4				A	A	D	D	D	E	G	F	F	F	F	F	F	F	E	D	D	D	A	A				46
5			A	A	C	C	C	F	B	E	A	E	B	B	E	A	E	B	F	C	C	C	A	A			44
6			A	D	C	G	E	C	E	C	E	C	E	E	C	E	C	E	C	E	G	C	D	A			42
7			A	D	C	E	B	F	C	E	C	E	A	A	E	C	E	C	F	B	E	C	D	A			40
8		A	A	D	F	C	F	C	E	A	E	A	E	E	A	E	A	E	C	F	C	F	D	A	A		38
9	A	C	C	E	B	E	C	E	A	E	B	E	B	B	E	B	E	A	E	C	E	B	E	C	C	A	36
10	A	D	C	F	E	C	E	A	E	A	E	C	E	E	C	E	A	E	A	E	C	E	F	C	D	A	34
11	A	C	C	F	A	E	C	E	B	E	C	E	A	A	E	C	E	B	E	C	E	A	F	C	C	A	32
12	A	D	C	F	E	C	E	A	E	C	E	C	E	E	C	E	C	E	A	E	C	E	F	C	D	A	30
13	A	D	C	F	B	E	A	E	B	E	A	E	B	B	E	A	E	B	E	A	E	B	F	C	D	A	28
14	A	D	C	F	B	E	A	E	B	E	A	E	B	B	E	A	E	B	E	A	E	B	F	C	D	A	26
15	A	D	C	F	E	C	E	A	E	C	E	C	E	E	C	E	C	E	A	E	C	E	F	C	D	A	24
16	A	C	C	F	A	E	C	E	B	E	C	E	A	A	E	C	E	B	E	C	E	A	F	C	C	A	22
17	A	D	C	F	E	C	E	A	E	A	E	C	E	E	C	E	A	E	A	E	C	E	F	C	D	A	20
18	A	C	C	E	B	E	C	E	A	E	B	E	B	B	E	B	E	A	E	C	E	B	E	C	C	A	18
19		A	A	D	F	C	F	C	E	A	E	A	E	E	A	E	A	E	C	F	C	F	D	A	A		16
20			A	D	C	E	B	F	C	E	C	E	A	A	E	C	E	C	F	B	E	C	D	A			14
21			A	D	C	G	E	C	E	C	E	C	E	E	C	E	C	E	C	E	G	C	D	A			12
22			A	A	C	C	C	F	B	E	A	E	B	B	E	A	E	B	F	C	C	C	A	A			10
23				A	A	D	D	D	E	F	F	F	F	F	F	F	G	E	D	D	D	A	A				8
24					A	A	A	A	C	C	C	C	C	C	C	C	C	C	A	A	A	A					6
25								A	C	D	C	D	D	D	D	C	D	C	A								4
26									A	A	A	A	A	A	A	A	A	A									2
	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	

		Number in Core	Cycle Loaded
A	GE9B-P8DWB348-12GZ-80U-145-T6	140	17
B	GE9B-P8DWB338-11GZ-80U-145-T6	40	17
C	GE9B-P8DWB348-12GZ-80U-145-T6	136	18
D	GE9B-P8DWB338-11GZ-80U-145-T6	48	18
E	GE11-P9HUB369-12GZ-100T-145-T6-2560	144	19
F	GE11-P9HUB374-13GZ-100T-145-T6-2559	46	19
G	GE9B-P8DWB348-12GZ-80U-145-T6	6	19
	Total	560	

[[ ]]
[[ ]]

ENCLOSURE 5

Global Nuclear Fuel Affidavit Certifying

Request for Withholding from Public Disclosure

Affidavit

I, John F. Schardt, being duly sworn, depose and state as follows:

(1) I am Manager, Fuel Technology and Design, Global Nuclear Fuel – Americas, L.L.C. ("GNF-A") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the attachment, "Additional Information Regarding the Cycle Specific SLMCPR for Oyster Creek Cycle 19," May 24, 2002.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4) and 2.790(a)(4) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information," and some portions also qualify under the narrower definition of "trade secret," within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of GNF-A, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future GNF-A customer–funded development plans and programs, of potential commercial value to GNF-A;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in (6) and (7) following. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology.

The development of the methods used in these analyses, along with the testing, development and approval of the supporting methodology was achieved at a significant cost, on the order of several million dollars, to GNF-A or its licensor.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit–making opportunities. The fuel design and licensing methodology is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC–approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GNF-A or its licensor.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed at Wilmington, North Carolina, this 24 day of May, 2002.

John F. Schardt

John F. Schardt
Global Nuclear Fuel – Americas, LLC