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10 CFR 50.90

August 27, 2004
2130-02-20203

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

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

Subject: Technical Specification Change Request No. 331 – Safety Limit Minimum Critical Power Ratio

Pursuant to 10 CFR 50.90 AmerGen Energy Company, LLC (AmerGen), hereby requests the following amendment to the Technical Specification, Appendix A of Operating License No. DPR-16 for Oyster Creek Generating Station (OCGS). This proposed change will revise Technical Specification (TS) Section 2.1.A. This section will be revised to incorporate revised Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the cycle specific analysis performed by Global Nuclear Fuel for OCGS, Cycle 20. This information is being submitted under unsworn declaration.

Information supporting this TS Change Request is contained in Attachment 1 to this letter, and the proposed marked up TS page and final TS page are contained in Attachments 2 and 3, respectively. Attachment 4 (letter from J. M. Downs (Global Nuclear Fuel) to R. Tropasso (Exelon Generation Company, LLC), dated July 2, 2004) specifies the new SLMCPRs for OCGS, Cycle 20. Attachment 4 contains information proprietary to Global Nuclear Fuel. Global Nuclear Fuel requests that the document be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4). An affidavit supporting this request is also contained in Attachment 4. Attachment 5 contains a non-proprietary version of the Global Nuclear Fuel document.

In order to support the upcoming refueling outage at OCGS, AmerGen requests approval of the proposed amendment by November 1, 2004. Once approved, this amendment shall be implemented within 60 days of issuance.

Additionally, there are no commitments contained within this letter.

A001

These proposed changes have been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91(b)(1), a copy of this TS 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.

If you have any questions or require additional information, please contact Tom Loomis at (610) 765-5510.

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

Respectfully,

8/27/2004
Executed on

C. N. Swenson
Vice President, Oyster Creek Generating Station

Attachments:
1- Licensee's Evaluation
2- Markup of Technical Specification Pages
3- Camera Ready Technical Specification Pages
4- Proprietary Global Nuclear Fuel Letter
5- Non-proprietary Version of Global Nuclear Fuel Letter

cc: S. J. Collins, Administrator, USNRC Region I
P. S. Tam, USNRC Senior Project Manager, Oyster Creek
R. J. Summers, USNRC Senior Resident Inspector, Oyster Creek
File No. 02079

ATTACHMENT 1

OYSTER CREEK GENERATING STATION

DOCKET NO. 50-219

LICENSE NO. DPR-16

Technical Specification Change Request No. 331 - Safety Limit Minimum Critical Power Ratio

ATTACHMENT 1 CONTENTS

SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (SLMCPR) CHANGE

1.0 DESCRIPTION

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

The proposed changes would revise the Operating License to 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 Generating Station (OCGS) Cycle 20, which includes the use of GE-9B and GE-11 fuel product lines. NRC approval of this change is requested by November 1, 2004 in order to allow the revised SLMCPR values to be implemented prior to restart from the upcoming OCGS outage.

2.0 PROPOSED CHANGE

The proposed change involves revising the SLMCPR values contained in Technical Specification (TS) 2.1.A for three, four and five recirculation loop operation. The SLMCPR value for four and five loop operation is being changed from 1.09 to 1.10. The SLMCPR value for three loop operation is being changed from 1.10 to 1.12.

Marked up TS page 2.1-1, showing the requested changes are provided in Attachment 2.

3.0 BACKGROUND

The proposed amendment involves revising the SLMCPR values contained in TS 2.1.A (page 2.1-1) from 1.10 to 1.12 for three recirculation loop operation and from 1.09 to 1.10 for both four and five recirculation loop operation. The revised SLMCPR values were determined for OCGS based on the reload core design for Cycle 20. The current SLMCPR values were determined in accordance with NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (GESTAR II), Amendment 25. Amendment 25 provides the methodology for determining the cycle specific MCPR safety limits. Amendment 25 was used in determining the Cycle 20 SLMCPR values, and it is intended to use Amendment 25 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)]. The SLMCPRs have been calculated using the approved methodology of NEDC-32601P-A. Furthermore, GNF has generically increased uncertainties used in the SLMCPR analysis to account for the potential impact of control blade shadow corrosion induced bow.

4.0 TECHNICAL ANALYSIS

The proposed change to Technical Specifications will revise TS 2.1.A to reflect the cycle specific analysis performed by Global Nuclear Fuel for OCGS Cycle 20, which includes the use of GE-9B and GE-11 fuel product lines.

The proposed SLMCPR values were determined in accordance with the NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), which incorporates Amendment 25. Amendment 25 provides the methodology for determining the cycle specific MCPR safety limits. The NRC safety evaluation approving Amendment 25 is contained in a letter from the NRC to General Electric Company, dated March 11, 1999. Future SLMCPRs determined in accordance with Amendment 25 will not need prior NRC approval for each cycle unless the value changes.

The SLMCPRs have also been calculated using the approved methodology of NEDC-32601P-A. Furthermore, GNF has generically increased uncertainties used in the SLMCPR analysis to account for the potential impact of control blade shadow corrosion induced bow.

The SLMCPR analysis establishes SLMCPR values that will ensure that at least 99.9% of all fuel rods in the core avoid transition boiling if the limit is not exceeded. 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 Cycle 20 are 1.12 (three loop operation) and 1.10 (for both four loop and five loop operation). Additional information regarding the cycle specific SLMCPR values for Oyster Creek Cycle 20 is contained in Attachment 4.

The analyses performed demonstrate the proposed change is acceptable since no fuel thermal limits or other licensing basis acceptance criteria are adversely affected.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

We have concluded that the proposed change to the OCGS Technical Specifications, which will revise TS 2.1.A, does not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92(c) is provided below.

1. Does the proposed amendment 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, 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), 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 at least 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The revised SLMCPR values developed in the revised analysis preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. 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. Neither the probability nor the consequences of fuel damage will be increased as a result of this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

Response: No.

The SLMCPR is a TS numerical value, designed to ensure that transition boiling does not occur in greater than 99.9% of all fuel rods in the core if the limit is not violated. The revised SLMCPR values are calculated using NRC approved methodologies discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), June, 2000, which incorporates Amendment 25.

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

3. <u>Does the proposed amendment involve a significant reduction in a margin of safety?</u>

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. The revised SLMCPR values are calculated using methodology discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-14 (GESTAR-II), June, 2000, which incorporates Amendment 25. The SLMCPR values ensure that at least 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. 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.

5.2 Applicable Regulatory Requirements/Criteria

Safety limits are required to be included in the Technical Specifications by 10 CFR 50.36. The SLMCPR ensures sufficient conservatism in the operating MCPR limit that during normal operation and during abnormal operational transients, at least 99.9% of all fuel rods in the core do not experience transition boiling considering the power distribution within the core and all uncertainties.

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.

6.0 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.

7.0 REFERENCES

a) NEDE-24011-P-A-14 (GESTAR-II), "General Electric Standard Application for Reactor Fuel", which incorporates Amendment 25.

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) Letter from J. M. Downs (Global Nuclear Fuel) to R. Tropasso (Exelon Generation Company, LLC), dated July 2, 2004 (Proprietary).

ATTACHMENT 2

OYSTER CREEK GENERATING STATION

DOCKET NO. 50-219

LICENSE NO. DPR-16

Technical Specification Change Request No. 331 - Safety Limit Minimum Critical Power Ratio

MARKED UP TECHNICAL SPECIFICATION PAGE

2.1-1

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:

1,10

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 for both four or five loop operation and 1.10 for three loop operation shall constitute violation of the fuel cladding integrity safety limit.

1,12

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.

ATTACHMENT 3

DOCKET NO. 50-219

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Technical Specification Change Request No. 331 - Safety Limit Minimum Critical Power Ratio

CAMERA-READY TECHNICAL SPECIFICATION

2.1-1

SECTION 2

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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 1.10 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.

ATTACHMENT 5

OYSTER CREEK GENERATING STATION

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Technical Specification Change Request No. 331 - Safety Limit Minimum Critical Power Ratio

NON-PROPRIETARY VERSION

Proprietary Information Notice

This document is the GNF non-proprietary version of the GNF proprietary report. From the GNF proprietary version, the information denoted as GNF proprietary (enclosed in double brackets) was deleted to generate this version.

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 Nos. 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, Glen A. Watford (GNF-A) to U. S. Nuclear Regulatory Commission Document Control Desk with attention to J. Donoghue (NRC), "Final Presentation Material for GEXL Presentation – February 11, 2002", FLN-2002-004, February 12, 2002.

Discussion

The Safety Limit Minimum Critical Power Ratio (SLMCPR) evaluations for the Oyster Creek Cycle 20 were performed using NRC approved methodology and uncertainties [1]. Table 1 summarizes the relevant input parameters and results of Cycle 20 and Cycle 19 cores. Additional information is provided in response to NRC questions related to similar submittals regarding changes in Technical Specification values of SLMCPR. Items that require a plant/cycle specific response are presented below.

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 impact of these parameters on the Oyster Creek Cycle 20 and Cycle 19 SLMCPR values is summarized in Table 1.

The core loading information for Oyster Creek Cycle 19 is provided in Figure 1. For comparison the core loading information for Oyster Creek Cycle 20 is provided in Figure 2. The impact of the fuel loading pattern differences on the calculated SLMCPR is correlated to the values of [[

[3]]]

[[

[3]]]

The uncontrolled bundle pin-by-pin power distributions were compared between the Oyster Creek Cycle 20 bundles and the Cycle 19 bundles. Pin-by-pin power distributions are characterized in terms of R-factors using the NRC approved methodology [2]. For the Oyster Creek Cycle 20 limiting case analyzed at EOC, [[

[3]]] the Oyster Creek Cycle 20 bundles have a more peaked power distribution than the bundles used for the Cycle 19 SLMCPR analysis.

As shown in Table 1, the SLMCPR for both four and five loop operation (FLO) is 1.10. 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 20.

The SLMCPR was calculated for Oyster Creek Cycle 20 using uncertainties that have been previously reviewed and approved by the NRC. These uncertainties are shown in Table 2a and described in Reference [1]. Where warranted, higher plant-cycle-specific uncertainties were used, as listed in Table 2b.

Attachment | **Additional Information Regarding the Cycle Specific SLMCPR for Oyster Creek Cycle 20** | **July 2, 2004**

Summary

The calculated 1.10 SLMCPR for four and five loop operation (FLO) and 1.12 3LO SLMCPR for Oyster Creek Cycle 20 are consistent with expectations [[

[3]]] these values are appropriate when the approved methodology given in NEDC-32601P-A is used.

Based on the information and discussion presented above, it is concluded that the calculated SLMCPR of 1.10 for FLO and 1.12 for 3LO are appropriate for the Oyster Creek Cycle 20 core..

Prepared by :

John P. Rea
Technical Program Manager
Global Nuclear Fuel - Americas

Verified by:

Anghel Enica
Technical Program Manager
Global Nuclear Fuel - Americas

Table 1
Comparison of the Oyster Creek Cycle 20 and Cycle 19 SLMCPR

QUANTITY, DESCRIPTION	Oyster Creek Cycle 19	Oyster Creek Cycle 20
Number of Bundles in Core	560	560
Limiting Cycle Exposure Point	EOC	EOC
Cycle Exposure at Limiting Point (MWd/STU)	10400	10500
Reload Fuel Type	GE11	GE11
Latest Reload Batch Fraction, %	37.2	30.0
Latest Reload Average Batch Weight % Enrichment	3.70	3.63
Core Fuel Fraction for GE11 (%)	33.9	63.9
Core Fuel Fraction for GE9B (%)	66.1	36.1
Core Average Weight % Enrichment	3.54	3.59
Core MCPR (for limiting rod pattern)	1.55	1.46
[[		(3)]]
[[		(3)]]
[[		(3)]]
Power distribution methodology	Revised NEDC-32601P-A	Revised NEDC-32601P-A
Power distribution uncertainty	GETAB NEDO-10958-A	GETAB NEDO-10958-A
Non-power distribution uncertainty	Revised NEDC-32601P-A	Revised NEDC-32601P-A
Calculated Safety Limit MCPR (FLO)	**1.09**	**1.10**
Calculated Safety Limit MCPR (3LO)	**1.10**	**1.12**

Table 2a

Standard Uncertainties

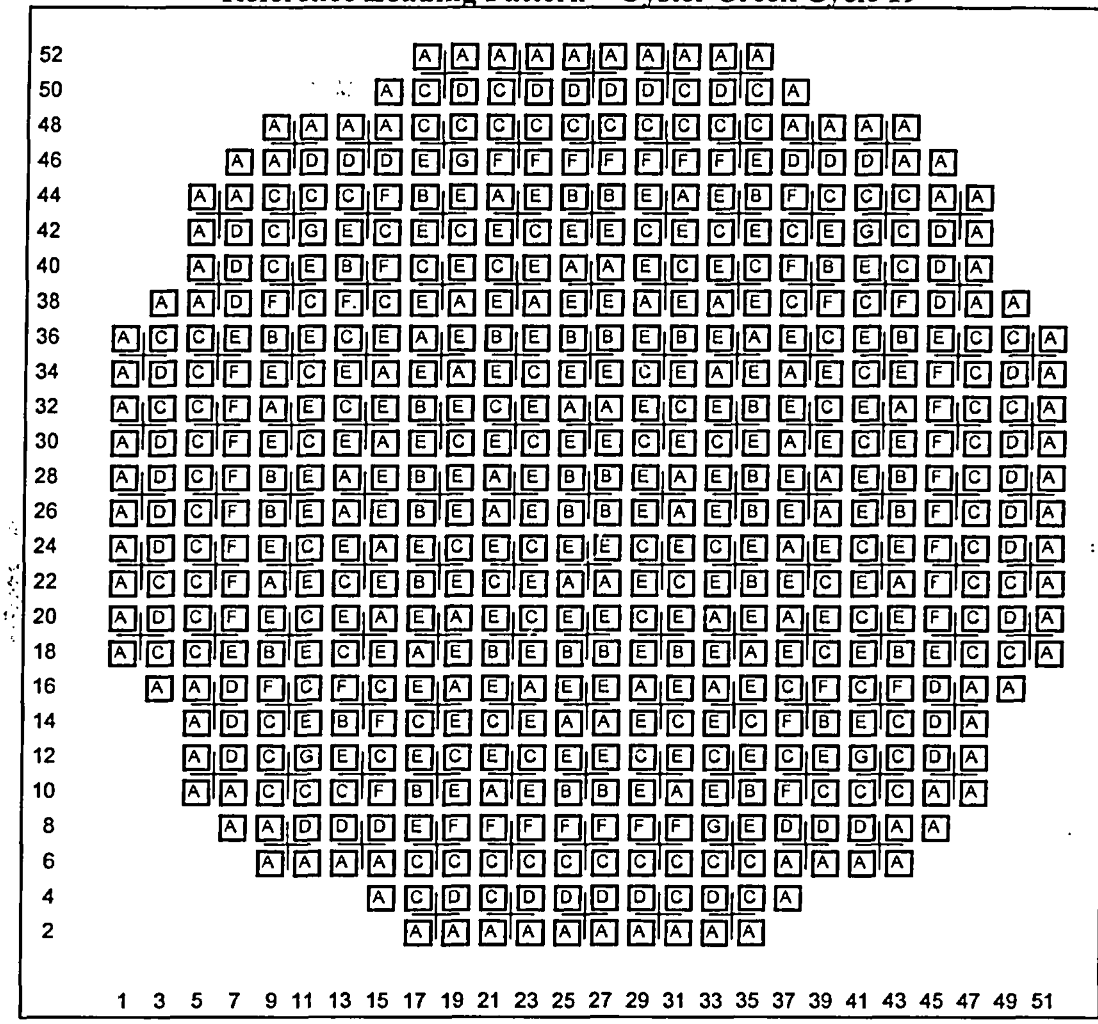
DESCRIPTION	Oyster Creek Cycle 19	Oyster Creek Cycle 20
Non-power Distribution Uncertainties	Revised NEDC-32601P-A	Revised NEDC-32601P-A
Core flow rate (derived from pressure drop)	2.5 FLO 6.0 3LO	2.5 FLO 6.0 3LO
Individual channel flow area	[[{3}]]	[[{3}]]
Individual channel friction factor	5.0	5.0
Friction factor multiplier	[[{3}]]	[[{3}]]
Reactor pressure	[[{3}]]	[[{3}]]
Core inlet temperature	0.2	0.2
Feedwater temperature	[[{3}]]	[[{3}]]
Feedwater flow rate	[[{3}]]	[[{3}]]
Power Distribution Uncertainties	GETAB NEDO-10958-A and Revised NEDC-32601P-A	GETAB NEDO-10958-A and Revised NEDC-32601P-A
GEXL R-factor	[[{3}]]	[[{3}]]
Random effective TIP reading	1.2 FLO 2.85 3LO	1.2 FLO 2.85 3LO
Systematic effective TIP reading	[[{3}]]	[[{3}]]
Integrated effective TIP reading	[[{3}]]	[[{3}]]
Bundle power	[[{3}]]	[[{3}]]
Effective total bundle power uncertainty	4.3	4.3

Table 2b

Exceptions to the Standard Uncertainties Used in Oyster Creek Cycle 20

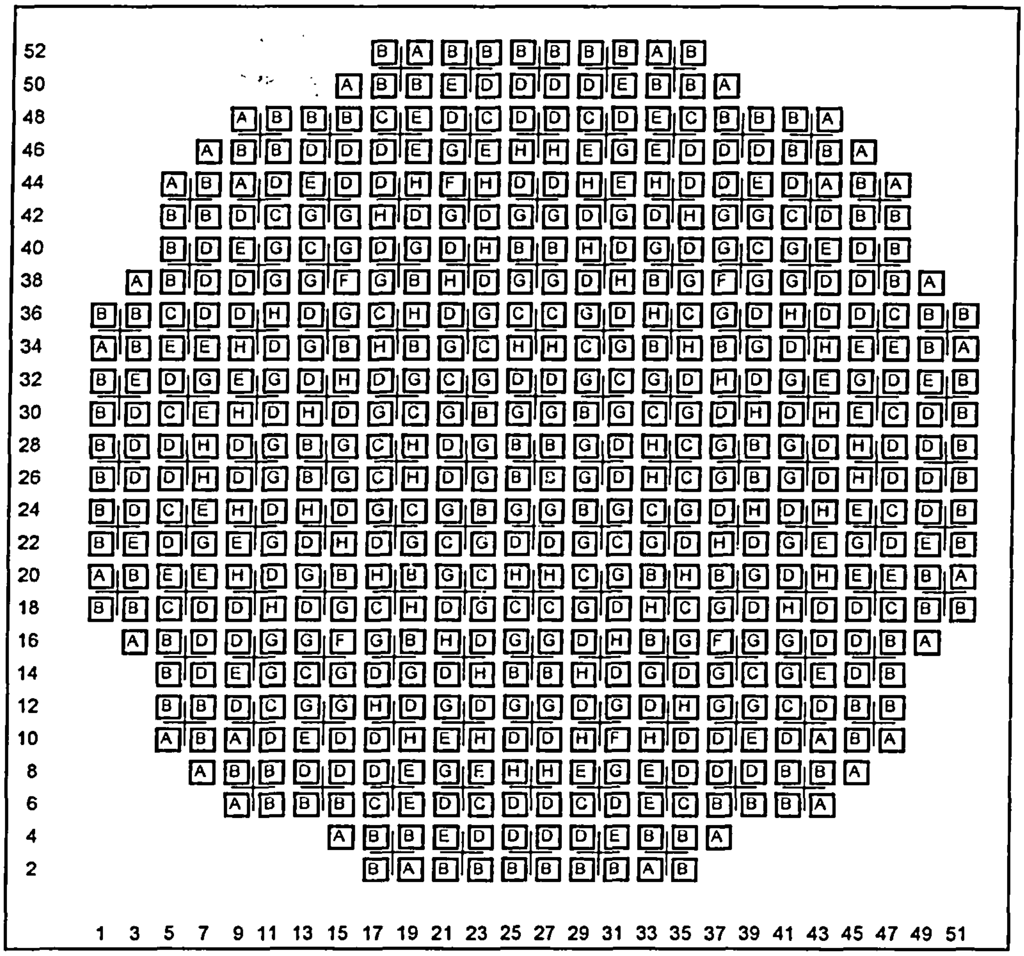
GEXL R-factor	[[{3}]]

Figure 1
Reference Loading Pattern – Oyster Creek Cycle 19



Code	Bundle Name	Number Loaded	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

Figure 2
Reference Loading Pattern – Oyster Creek Cycle 20



Code	Bundle Name	Number Loaded	Cycle Loaded
A	GE9B-P8DWB348-12GZ-80U-145-T6	32	17
B	GE9B-P8DWB348-12GZ-80U-145-T6	116	18
C	GE9B-P8DWB338-11GZ-80U-145-T6	48	18
D	GE11-P9HUB369-12GZ-100T-145-T6-2560	144	19
E	GE11-P9HUB374-13GZ-100T-145-T6-2559	46	19
F	GE9B-P8DWB348-12GZ-80U-145-T6	6	19
G	GE11-P9HUB363-12GZ-100T-145-T6-2817	104	20
H	GE11-P9HUB364-14GZ-100T-145-T6-2818	64	20