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10 CFR 50.90

December 14, 2005

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Limerick Generating Station, Unit 1
Facility Operating License No. NPF-39
NRC Docket No. 50-352

SUBJECT: License Amendment Request
Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) Change

Dear Sir/Madam:

Pursuant to 10 CFR 50.90 Exelon Generation Company, LLC (Exelon), hereby requests the following amendment to the Technical Specifications (TS), Appendix A of Operating License No. NPF-39 for Limerick Generating Station (LGS), Unit 1. This proposed change will revise Technical Specification (TS) Section 2.1. This Section will be revised to incorporate a revised Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) due to the cycle specific analysis performed by Global Nuclear Fuel for LGS, Unit 1, Cycle 12. The two-loop SLMCPR will not change.

In order to support the upcoming refueling outage at LGS, Unit 1, Exelon requests approval of the proposed amendment by March 1, 2006. Once approved, this amendment shall be implemented within 30 days of issuance. Additionally, there are no commitments contained within this letter.

This proposed change has been reviewed by the Plant Operations Review Committee, and approved by the Nuclear Safety Review Board.

Information supporting this License Amendment Request is contained in Enclosure 1 to this letter, and the proposed marked up pages and final camera ready pages are contained in Attachments 2 and 3, respectively. Enclosure 4 (letter from M. J. Mneimneh {Global Nuclear Fuel} to J. Tusar {Exelon Generation Company, LLC}, dated November 9, 2005) specifies the new SLO SLMCPR for LGS, Unit 1. Enclosure 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 2.390(a)(4). An affidavit supporting this request is also contained in Enclosure 4. Enclosure 5 contains a non-proprietary version of the Global Nuclear Fuel document.

We are notifying the Commonwealth of Pennsylvania 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 Tom Loomis at (610) 765-5510.

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

Respectfully,

12/14/05
Executed On

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Enclosures: 1-Evaluation of Proposed Change
2-Markup of Technical Specification Pages
3-Camera Ready Technical Specification Pages
4-Proprietary Version of Global Nuclear Fuel Letter
5-Non-Proprietary Version of Global Nuclear Fuel Letter

cc: S. J. Collins, Administrator, USNRC Region I
S. Hansell, USNRC Senior Resident Inspector, LGS
G. Wunder, Project Manager, USNRC
R. R. Janati, Commonwealth of Pennsylvania

ENCLOSURE 1

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO. 50-352

LICENSE NO. NPF-39

LICENSE AMENDMENT REQUEST

EVALUATION OF PROPOSED CHANGE

ENCLOSURE 1 CONTENTS

SUBJECT: Revision to Single Loop Operation Safety Limit Minimum Critical Power Ratio

1.0 DESCRIPTION

Exelon Generation Company, LLC (Exelon) Licensee under Facility Operating License No. NPF-39 for Limerick Generating Station (LGS), Unit 1, requests that the Technical Specifications (TS) contained in Appendix A to the Operating License be amended to revise TS 2.1 to reflect a change in the Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) due to the cycle specific analysis performed by Global Nuclear Fuel for LGS, Unit 1, Cycle 12. The two-loop SLMCPR will not change.

The marked up Technical Specification page and camera ready Technical Specification page are contained in Enclosures 2 and 3, respectively. Also included in Enclosures 2 and 3 are the associated Bases changes, which are being supplied to you for your information. Enclosure 4 (letter from M. J. Mneimneh (Global Nuclear Fuel) to J. Tusar (Exelon Generation Company, LLC), dated November 9, 2005) specifies the new SLO SLMCPR for LGS, Unit 1, Cycle 12.

2.0 PROPOSED CHANGE

This proposed change will revise Technical Specification (TS) Section 2.1. This Section will be revised to incorporate a revised Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) due to the cycle specific analysis performed by Global Nuclear Fuel for LGS, Unit 1, Cycle 12. The new SLO SLMCPR at LGS, Unit 1, Cycle 12 is 1.09. The two-loop operation SLMCPR will remain the same (1.07) as shown in Enclosure 4. Additional information regarding the 1.09 cycle specific SLO SLMCPR for LGS, Unit 1 Cycle 12 is contained in the Enclosure 4 letter.

3.0 BACKGROUND

The proposed change involves revising the Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) value contained in TS 2.1 for single recirculation loop operation. The SLO SLMCPR value is being revised for LGS, Unit 1 based on the reload core design for Cycle 12, which uses the GE-14 fuel product line. GE-14 fuel has previously been loaded into the Limerick Generating Station, Unit 1 core. The SLO SLMCPR has been determined in accordance with NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005, which includes Amendment 25. Amendment 25 provides the methodology for determining the cycle specific MCPR safety limits that replace the former generic fuel type dependent values. Amendment 25 is being used for determining the upcoming Cycle 12 SLO SLMCPR. Future SLO SLMCPRs determined in accordance with Amendment 25 will not need prior NRC approval for each cycle unless the TS 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 (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 GE-14 fuel to be in compliance with Amendment 22 included in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005.

4.0 TECHNICAL ANALYSIS

The proposed TS change will revise TS 2.1 to reflect the cycle specific analysis performed by Global Nuclear Fuel for LGS, Unit 1, Cycle 12, which includes the use of the GE-14 fuel product line.

The new SLO SLMCPR is calculated using NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005, which includes Amendment 25. Amendment 25 is being used for determining the upcoming Cycle 12 SLO SLMCPR. Future SLO SLMCPRs determined in accordance with Amendment 25 will not need prior NRC approval for each cycle unless a TS 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 GE-14 fuel to be in compliance with Amendment 22 to "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005.

The SLO SLMCPR analysis establishes SLO 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 SLO SLMCPRs 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 Cycle 11 calculated SLO SLMCPR is 1.084 and the Cycle 12 calculated SLO SLMCPR is 1.089. These results indicate that the calculated SLO SLMCPR results for both Cycles 11 and 12 differ by +0.005. This +0.005 increase in the SLO SLMCPR is attributed to the statistical variations in the Monte Carlo analysis. The increase in SLO SLMCPR of 0.01 is attributed to rounding down the Cycle 11 result to 1.08, and rounding up the Cycle 12 result to 1.09. The new SLO SLMCPR at LGS, Unit 1, Cycle 12 is 1.09. The two-loop operation SLMCPR will not change as shown in Enclosure 4. Additional information regarding the cycle specific SLO SLMCPR for LGS, Unit 1 Cycle 12 is contained in the Enclosure 4 letter.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Exelon 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 Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) for incorporation into the Technical Specifications (TS), and its use to determine cycle specific thermal limits, has been performed using the methodology discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005, which includes Amendment 25. Amendment 25 was approved by the NRC in a March 11, 1999 safety evaluation report.

   The basis of the SLO 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 SLO SLMCPR preserves the existing margin to transition boiling. The GE-14 fuel is in compliance with Amendment 22 to "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September 2005, which provides the fuel licensing acceptance criteria. The probability of fuel damage will not be increased as a result of this change. Therefore, the proposed TS 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 SLO SLMCPR is a TS numerical value, calculated 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 SLO SLMCPR is calculated using NRC approved methodology discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U.S. Supplement, NEDE-24011-P-A-15-US, September 2005, which includes Amendment 25. Additionally, the GE-14 fuel is in compliance with Amendment 22 to "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005, which provides the fuel licensing acceptance criteria. The SLO 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.

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 SLO SLMCPR, which includes the use of GE-14 fuel. The new SLO SLMCPR is calculated using methodology

discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-15 (GESTAR-II), and U. S. Supplement, NEDE-24011-P-A-15-US, September, 2005, which includes Amendment 25. The SLO SLMCPR ensures 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. Therefore, the proposed TS change will not involve a significant reduction in the margin of safety previously approved by the NRC.

Based on the above, Exelon Generation Company, LLC, 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 SLO SLMCPR ensures sufficient conservatism in the operating SLO 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.

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, "Standards for Protection Against Radiation," 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, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, Paragraph (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-15 (GESTAR-II), "General Electric Standard Application for Reactor Fuel", and U.S. Supplement, NEDE-24011-P-A-15-US, September 2005, which includes 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) NEDC-32601P-A, Methodology and Uncertainties for Safety Limit MCPR Evaluations.

d) NEDC-32694P-A, Power Distribution Uncertainties for Safety Limit MCPR Evaluation.

e) Letter from M. J. Mneimneh (Global Nuclear Fuel) to J. Tusar (Exelon Generation Company, LLC), dated November 9, 2005.

ENCLOSURE 2

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO. 50-352

LICENSE NO. NPF-39

LICENSE AMENDMENT REQUEST

MARKUP TECHNICAL SPECIFICATION PAGES

UNIT 1

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

1.09

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 for two recirculation loop operation and shall not be less than 1.08 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 for two recirculation loop operation or less than 1.08 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

1.09

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

1.09

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/h, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/h. 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 THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

ENCLOSURE 3

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO. 50-352

LICENSE NO. NPF-39

LICENSE AMENDMENT REQUEST

CAMERA READY TECHNICAL SPECIFICATION PAGES

UNIT 1

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 for two recirculation loop operation and shall not be less than 1.09 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 for two recirculation loop operation or less than 1.09 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.09 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.09 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/h, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/h. 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 THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

ENCLOSURE 5

LIMERICK GENERATING STATION
UNIT 1

Docket No. 50-352

License No. NPF-39

LICENSE AMENDMENT REQUEST

NON-PROPRIETARY VERSION OF GLOBAL NUCLEAR FUEL LETTER

Proprietary Information Notice

This document is the GNF non-proprietary version of the GNF proprietary report. From the GNF proprietary version, the information noted 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 R. Pulsifer (NRC), "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies", FLN-2001-016, September 24, 2001.

[5] Letter, Glen A. Watford (GNF-A) to U. S. Nuclear Regulatory Commission Document Control Desk with attention to J. Donoghue (NRC), "Confirmation of the Applicability of the GEXL14 Correlation and Associated R-Factor Methodology for Calculating SLMCPR Values in Cores Containing GE14 Fuel", FLN-2001-017, October 1, 2001

[6] Letter, Jason S. Post (GE Energy) to U.S. Nuclear Regulatory Commission Document Control Desk, "Part 21 Reportable Condition and 60-Day Interim Report Notification: Non-conservative SLMCPR", MFN-04-081, August 24, 2004.

[7] 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 Limerick 1 Cycle 12 were performed using NRC approved methodology and uncertainties [1]. Table 1 summarizes the relevant input parameters and results of Cycle 12 and Cycle 11 cores. Additional information is provided in response to NRC questions related to similar submittals regarding changes in Technical Specification values of SLMCPR. NRC questions pertaining to how GE14 applications satisfy the conditions of the NRC SER[1] have been addressed in Reference [4]. Other generically applicable questions related to application of the GEXL14 correlation, and to the applicable range for the R-factor methodology, are addressed in Reference [5]. Items that require a plant/cycle specific response are presented below.

Previously, the SLMCPR was calculated on the upper boundary of the power/flow operating map only at 100% flow / 100% power (rated flow/rated power), which had been shown in NEDC-32601P-A to result in conservative SLMCPR evaluation values using the same control rod pattern used for rated flow/rated power evaluations. Recent evaluations for BWR plants fueled by GNF fuel bundle designs determined that limiting control blade patterns developed for less than rated flow at rated power condition sometimes yield more limiting bundle-by-bundle MCPR distributions and/or more limiting bundle axial power shapes than the limiting control blade patterns developed for a rated flow/rated power SLMCPR evaluation, as reported in Reference [6]. Therefore, to conservatively account for operation at lower flow/rated power conditions, SLMCPR evaluations were also performed at the lowest core flow rate (81% rated flow) at rated power condition for the same exposure points that were previously calculated for the rated flow/rated power evaluations. The results for Limerick 1 Cycle 12 at the lower flow condition bounded those at the rated core flow condition.

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 Limerick 1 Cycle 12 and Cycle 11 SLMCPR values is summarized in Table 1.

The core loading information for Limerick 1 Cycle 11 is provided in Figure 1. For comparison the core loading information for Limerick 1 Cycle 12 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 Limerick 1 Cycle 12 bundles and the Cycle 11 bundles. Pin-by-pin power distributions are characterized in terms of R-factors using the NRC approved methodology [2]. For the Limerick 1 Cycle 12 limiting case analyzed at EOC_MELLLA, [[

{3}]] the Limerick 1 Cycle 12 bundles have a more peaked power distribution than the bundles used for the Cycle 11 SLMCPR analysis.

The SLMCPR was calculated for Limerick 1 Cycle 12 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.

Table 1 summarizes the relevant input parameters and results of Cycle 11 at rated flow and power and for Cycle 12 evaluated at the limiting condition of 81% rated flow/rated power. The SLMCPR values were calculated for Limerick 1 using uncertainties that have been previously reviewed and approved by the NRC as listed in Table 2a and described in Reference [1] and where warranted, higher plant-cycle-specific uncertainties as listed in Table 2b. In addition to using a [[{3}]] consistent with current GNF fuel operation, for the lower flow evaluations, the Core Flow Rate and Random effective TIP reading uncertainties were [[

{3}]] for the Limerick 1 low flow Cycle 12 evaluation.

These calculations use the GEXL14 correlation for GE14 fuel. [[

{3}]]

For single loop operations (SLO) the calculated safety limit MCPR for the limiting case is 1.09 as determined by specific calculations for Limerick 1 Cycle 12 at EOC_MELLLA. The DLO and SLO SLMCPR values calculated for Limerick 1 Cycle 12 are shown in Table 1.

Summary

The calculated 1.07 SLMCPR and 1.09 SLO SLMCPR for Limerick 1 Cycle 12 are consistent with expectations [[

{3}]] these values are appropriate when the approved methodology and the reduced uncertainties given in NEDC-32601P-A and NEDC-32694P-A are used.

Based on the information and discussion presented above, it is concluded that the calculated SLMCPR of 1.07 and 1.09 for SLO are appropriate for the Limerick 1 Cycle 12 core.

Prepared by :

Technical Program Manager
Global Nuclear Fuel - Americas

Verified by:

Technical Program Manager
Global Nuclear Fuel - Americas

Table 1
Comparison of the Limerick 1 Cycle 12 and Cycle 11 SLMCPR

QUANTITY, DESCRIPTION	**Limerick 1 Cycle 11**	**Limerick 1 Cycle 12**
Number of Bundles in Core	764	764
Limiting Cycle Exposure Point	13600	EOC_MELLLA
Cycle Exposure at Limiting Point (MWd/STU)	13600	11400
Reload Fuel Type	GE14	GE14
Latest Reload Batch Fraction, %	34.6	36.1
Latest Reload Average Batch Weight % Enrichment	4.16	4.03
Core Fuel Fraction for GE14 (%)	71.2	100.0
Core Fuel Fraction for GE13 (%)	28.8	0.0
Core Average Weight % Enrichment	4.17	4.12
Core MCPR (for limiting rod pattern)	1.34	1.34
[[		{3}]]
[[		{3}]]
[[		{3}]]
Power distribution methodology	Revised NEDC-32601P-A	Revised NEDC-32601P-A
Power distribution uncertainty	Reduced NEDC-32694P-A	Reduced NEDC-32694P-A
Non-power distribution uncertainty	Revised NEDC-32601P-A	Revised NEDC-32601P-A
Calculated Safety Limit MCPR (DLO)	**1.07**	**1.07**
Calculated Safety Limit MCPR (SLO)	**1.08**	**1.09**

Table 2a

Standard Uncertainties

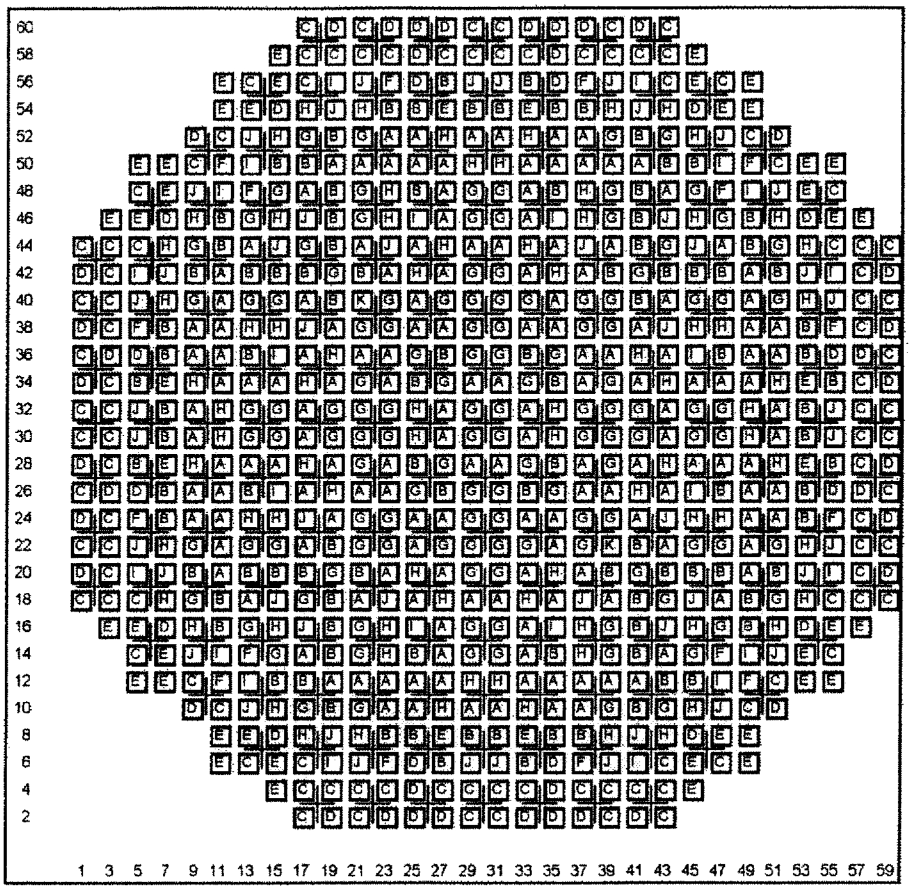
DESCRIPTION	Limerick 1 Cycle 11	Limerick 1 Cycle 12
Non-power Distribution Uncertainties	Revised NEDC-32601P-A	Revised NEDC-32601P-A
Core flow rate (derived from pressure drop)	2.5 DLO 6.0 SLO	2.5 DLO 6.0 SLO
Individual channel flow area	[[{3}]]	[[{3}]]
Individual channel friction factor	5.0	5.0
Friction 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	Reduced NEDC-32694P-A	Reduced NEDC-32694P-A
GEXL R-factor	[[{3}]]	[[{3}]]
Random effective TIP reading	1.2 DLO 2.85 SLO	1.2 DLO 2.85 SLO
Systematic effective TIP reading	[[{3}]]	[[{3}]]
Integrated effective TIP reading	[[{3}]]	[[{3}]]
Bundle power	[[{3}]]	[[{3}]]
Effective total bundle power uncertainty	[[{3}]]	[[{3}]]

Table 2b

Exceptions to the Standard Uncertainties Used in Limerick 1 Cycle 12

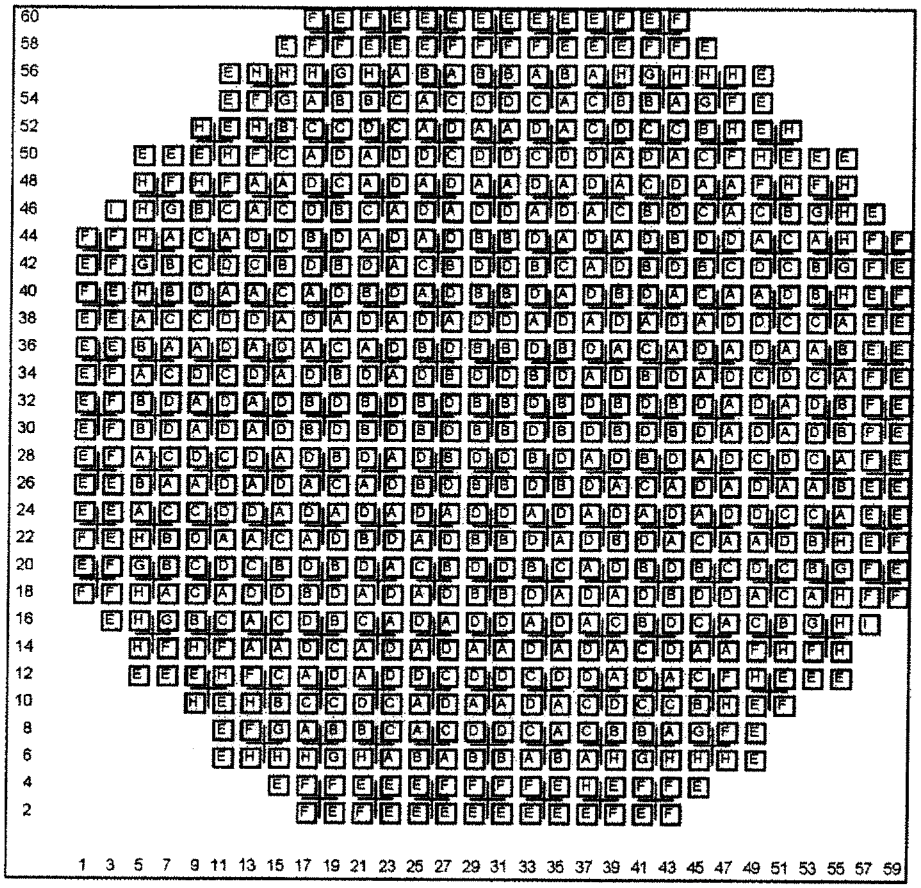
Core flow rate (DLO analysis only)	[[{3}]]
GEXL R-factor	[[{3}]]
Random effective TIP reading	[[{3}]]

Figure 1
Reference Loading Pattern – Limerick 1 Cycle 11



Code	Bundle Name	Number Loaded	Cycle Loaded
A	GE14-P10CNAB417-15GZ-100T-150-T6-2594	160	11
B	GE14-P10CNAB414-14GZ-100T-150-T6-2690	104	11
C	GE13-P9CTB417-13GZ-100T-146-T6-3833	100	9
D	GE13-P9CTB417-11GZ-100T-146-T6-3834	56	9
E	GE13-P9CTB417-13GZ-100T-146-T6-3833	48	9
F	GE13-P9CTB417-11GZ-100T-146-T6-3834	16	9
G	GE14-P10CNAB417-7G8.0/8G7.0-100T-150-T6-2529	126	10
H	GE14-P10CNAB417-13GZ-100T-150-T6-2530	80	10
I	GE14-P10CNAB417-7G8.0/8G7.0-100T-150-T6-2529	24	10
J	GE14-P10CNAB417-13GZ-100T-150-T6-2530	48	10
K	GE14-P10CNAB417-7G8.0/8G7.0-80U45R-150-T6-2532	2	10

Figure 2
Reference Loading Pattern – Limerick 1 Cycle 12



Code	Bundle Name	Number Loaded	Cycle Loaded
A	GE14-P10CNAB417-15GZ-100T-150-T6-2594	160	11
B	GE14-P10CNAB414-14GZ-100T-150-T6-2690	104	11
C	GE14-P10CNAB403-14GZ-120T-150-T6-2882	84	12
D	GE14-P10CNAB403-15GZ-120T-150-T6-2883	192	12
E	GE14-P10CNAB417-7G8.0/8G7.0-100T-150-T6-2529	93	10
F	GE14-P10CNAB417-13GZ-100T-150-T6-2530	65	10
G	GE14-P10CNAB417-7G8.0/8G7.0-100T-150-T6-2529	16	10
H	GE14-P10CNAB417-13GZ-100T-150-T6-2530	48	10
I	GE14-P10CNAB417-7G8.0/8G7.0-80U45R-150-T6-2532	2	10