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Exelon Nuclear

10 CFR 50.90

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

December 15, 2010

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Limerick Generating Station, Unit 2
Facility Operating License No. NPF-85
NRC Docket No. 50-353

Subject: License Amendment Request - Safety Limit Minimum Critical Power Ratio Change

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (Exelon) requests a proposed change to modify Technical Specification (TS) 2.1 ("Safety Limits"). Specifically, this change incorporates revised Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the cycle specific analysis performed by Global Nuclear Fuel for Limerick Generating Station (LGS), Unit 2, Cycle 12.

The proposed changes have been reviewed by the Limerick Generating Station Plant Operations Review Committee, and approved by the Nuclear Safety Review Board in accordance with the requirements of the Exelon Quality Assurance Program.

In order to support the upcoming refueling outage at LGS, Unit 2, Exelon requests approval of the proposed amendment by March 15, 2011. Once approved, this amendment shall be implemented within 30 days of issuance. Additionally, there are no commitments contained within this letter.

Attachment 1 contains the evaluation of the proposed changes. Attachments 2 and 3 provide the marked up TS and Bases pages and the retyped TS and Bases pages, respectively.

Attachment 4 (letter from J. M. Downs (Global Nuclear Fuel) to J. Tusar (Exelon Generation Company, LLC), dated November 23, 2010) specifies the new SLMCPRs for LGS, Unit 2, Cycle 12. Attachment 4 contains information proprietary to Global Nuclear Fuel. Global Nuclear Fuel

**Attachment 4 transmitted herewith contains Proprietary Information.
When separated from attachments, this document is decontrolled.**

requests that the document be withheld from public disclosure in accordance with 10 CFR 2.390(b)(4). Attachment 5 contains a non-proprietary version of the Global Nuclear Fuel document. An affidavit supporting this request is also contained in Attachment 5. Attachment 6 contains the power/flow maps for Cycles 11 and 12.

In accordance with 10 CFR 50.91, Exelon is notifying the State of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th of December 2010.

Respectfully,

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

Attachments:
1) Evaluation of Proposed Changes
2) Markup of Technical Specifications and Bases Pages
3) Retyped Technical Specifications and Bases Pages
4 Proprietary Version of Global Nuclear Fuel Letter
5) Affidavit and Non-Proprietary Version of Global Nuclear Fuel Letter
6) Power/Flow Maps for Cycles 11 and 12

cc: USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, LGS
USNRC Project Manager, LGS
R. R. Janati, Commonwealth of Pennsylvania

Attachment 1

Limerick Generating Station (LGS), Unit 2

Facility Operating License No. NPF-85

Evaluation of Proposed Changes

ATTACHMENT 1 CONTENTS

SUBJECT: Safety Limit Minimum Critical Power Ratio (SLMCPR) Change

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- 4.2 Precedents
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1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Facility Operating License No. NPF-85 for Limerick Generating Station (LGS), Unit 2.

The proposed change modifies Technical Specification (TS) 2.1 ("Safety Limits"). Specifically, this change incorporates revised Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the cycle specific analysis performed by Global Nuclear Fuel for LGS, Unit 2, Cycle 12.

2.0 DETAILED DESCRIPTION

The proposed change involves revising the SLMCPRs contained in TS 2.1 for two recirculation loop operation and single recirculation loop operation. The SLMCPR value for two-loop operation is being changed from ≥ 1.07 to ≥ 1.09. The SLMCPR value for single-loop operation is being changed from ≥ 1.09 to ≥ 1.12.

Marked up TS page 2-1 and Bases page B 2-1 showing the requested changes are provided in Attachment 2.

3.0 TECHNICAL EVALUATION

The proposed TS change will revise the SLMCPRs contained in TS 2.1 for two recirculation loop operation and single recirculation loop operation to reflect the changes in the cycle specific analysis performed by Global Nuclear Fuel for LGS, Unit 2, Cycle 12.

The new SLMCPRs are calculated using NRC-approved methodology described in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17. A listing of the associated NRC-approved methodologies for calculating the SLMCPRs is provided in Section 1.0 ("Methodology") of Attachment 4.

The SLMCPR analysis establishes SLMCPR values that will ensure 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 if the limit is not violated. The SLMCPRs are calculated to include cycle specific parameters and, in general, are dominated by two key parameters: 1) flatness of the core bundle-by-bundle MCPR distribution, and 2) flatness of the bundle pin-by-pin power/R-Factor distribution. Information to support the cycle specific SLMCPRs is included in Attachment 4. That attachment summarizes the methodology, inputs, and results for the change in the SLMCPRs. The LGS, Unit 2, Cycle 12 core will consist of GE14 and GNF2 fuel types.

Attachment 6 contains the power/flow maps for Cycles 11 and 12 (draft). A Measurement Uncertainty Recapture (MUR) power uprate is planned for implementation at LGS, Unit 2 starting with Cycle 12. A final power to flow map for Cycle 12 is under development. The revised Cycle 12 SLMCPRs were calculated at the MUR power level.

No plant hardware or operational changes are required with this proposed change.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical specifications," paragraph (c)(1), requires that power reactor facility TS include safety limits for process variables that protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The fuel cladding integrity SLMCPR is established to assure that at least 99.9% of the fuel rods in the core do not experience transition boiling during normal operation and abnormal operating transients. Thus, the SLMCPR is required to be contained in TS.

4.2 Precedents

The NRC has approved similar SLMCPR changes for a number of plants:

1) Letter from M. H. Chernoff (U.S. Nuclear Regulatory Commission) to K. W. Singer (Tennessee Valley Authority), "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Cycle-Specific Safety Limit Minimum Critical Power Ratio (TAC NO. MD1721) (TS-455)," dated February 6, 2007

2) Letter from J. Wiebe (U.S. Nuclear Regulatory Commission) to C. Pardee (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments RE: Safety Limit Minimum Critical Power Ratio (TAC NOS. MD7374 and MD7375)," dated February 28, 2008

3) Letter from J. Kim (U.S. Nuclear Regulatory Commission) to Site Vice President (Entergy Nuclear Operations, Inc.), "Pilgrim Nuclear Power Station – Issuance of Amendment RE: Technical Specification Change Concerning Safety Limit Minimum Critical Power Ratio (TAC NO. ME0241)," dated March 26, 2009

4) Letter from C. Lyon (U.S. Nuclear Regulatory Commission) to Vice President, Operations (Entergy Operations, Inc.), "Grand Gulf Nuclear Station, Unit 1 - Issuance of Amendment RE: Change to the Minimum Critical Power Ratio Safety Limit (TAC NO. ME2474)," dated March 25, 2010

5) Letter from J. D. Hughey (U.S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Generation Company, LLC), "Peach Bottom Atomic Power Station, Unit 2 – Issuance of Amendment RE: Safety Limit Minimum Critical Power Ratio Value Change (TAC NO. ME3994)," dated September 28, 2010

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (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 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 Ratios (SLMCPRs) for incorporation into the Technical Specifications (TS), and their use to determine cycle specific thermal limits, has been performed using the methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17.

   The basis of the SLMCPR calculation is to ensure 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 if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling.

   The MCPR safety limit is reevaluated for each reload using NRC-approved methodologies. The analyses for Limerick Generating Station (LGS), Unit 2, Cycle 12 have concluded that a two loop MCPR safety limit of ≥ 1.09, based on the application of Global Nuclear Fuel's NRC-approved MCPR safety limit methodology, will ensure that this acceptance criterion is met. For single-loop operation, a MCPR safety limit of ≥ 1.12 also ensures that this acceptance criterion is met. The MCPR operating limits are presented and controlled in accordance with the LGS, Unit 2 Core Operating Limits Report (COLR).

   The requested TS changes do not involve any plant modifications or operational changes that could affect system reliability or performance or that could affect the probability of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigating systems, and do not introduce any new accident initiation mechanisms. Therefore, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The SLMCPR is a TS numerical value, calculated to ensure 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 if the limit is not violated. The new SLMCPRs are calculated using NRC-approved methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17. The proposed changes do not involve any new modes of operation or any plant modifications. The proposed revised MCPR safety limits have been shown to be acceptable for Cycle 12 operation. The core operating limits will continue to be developed using NRC-approved methods. The proposed MCPR safety limits or methods for establishing the core operating limits do not result in the creation of any new precursors to an accident. Therefore, the

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

3. Does the proposed amendment 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 SLMCPRs. The new SLMCPRs are calculated using methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17. The SLMCPRs ensure 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 if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do 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 does not involve a 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.

4.4 Conclusions

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.

5.0 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.

6.0 REFERENCES

1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 17.

ATTACHMENT 2

Markup of Technical Specifications and Bases Pages

Revised Pages

TS 2-1
Bases B 2-1

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 1.09 for two recirculation loop operation and shall not be less than 1.09 1.12 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 1.09 for two recirculation loop operation or less than 1.09 1.12 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: OPERATION 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 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 principle 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.

1.12

1.09

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/hr, 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/hr. 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.

ATTACHMENT 3

Retyped Technical Specifications and Bases Pages

Revised Pages

TS 2-1
Bases B 2-1

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 1.09 for two recirculation loop operation and shall not be less than 1.12 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 1.09 for two recirculation loop operation or less than 1.12 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: OPERATION 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 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 principle 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 1.09 for two recirculation loop operation and 1.12 for single recirculation loop operation. MCPR greater than 1.09 for two recirculation loop operation and 1.12 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/hr, 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/hr. 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.

ATTACHMENT 5

Affidavit and Non-Proprietary Version of Global Nuclear Fuel Letter

Global Nuclear Fuel - Americas LLC

AFFIDAVIT

I, Anthony P. Reese, state as follows:

(1) I am the Manager, Reload Design & Analysis, of Global Nuclear Fuel – Americas, LLC (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 GNF-A proprietary report, GNF-0000-0125-7436-R0-P, *GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR, Limerick 2 C12*, Class III, (GNF-A Proprietary Information), dated November 2010. GNF-A proprietary information in GNF-0000-0125-7436-R0-P is identified by a dark red dotted underline inside double square brackets. [[This sentence is an example.[3]]] Figures and large equation objects containing GNF-A proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation [3] refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination.

(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 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies 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, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that 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 GNF-A and/or other companies.

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

c. Information that reveals aspects of past, present, or future GNF-A customer-funded development plans and programs, that may include potential products of GNF-A.

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

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

(8) The information identified in paragraph (2) above is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology for the Boiling Water Reactor (BWR). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GNF-A. The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GNF-A asset.

(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. 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 on this 23rd day of November, 2010

Anthony P. Reese
Manager, Reload Design & Analysis
Global Nuclear Fuel - Americas LLC

November 2010

GNF-0000-0125-7436-R0-NP

eDRF Section: 0000-0125-7436-R0

GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR

Limerick 2 Cycle 12

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.

Important Notice Regarding Contents of this Report
Please Read Carefully

The information contained in this document is furnished solely for the purpose(s) stated in the transmittal letter. The only undertakings of GNF-A with respect to information in this document are contained in contracts between GNF-A and its customers or participating utilities, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GNF-A makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

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TABLE 6. CRITICAL POWER UNCERTAINTIES 25

1.0 Methodology

GNF performs Safety Limit Minimum Critical Power Ratio (SLMCPR) calculation in accordance to NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Revision 17) using the following NRC-approved methodologies and uncertainties:

- NEDC-32601P-A "Methodology and Uncertainties for Safety Limit MCPR Evaluations" (August 1999).
- NEDC-32694P-A "Power Distribution Uncertainties for Safety Limit MCPR Evaluations" (August 1999).
- NEDC-32505P-A "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel" (Revision 1, July 1999).
- NEDO-10958-A "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application" (January 1977).

Table 2 identifies the actual methodologies used for the Limerick 2 Cycle 11 and the Cycle 12 SLMCPR calculations.

2.0 Discussion

In this discussion, the TLO nomenclature is used for two recirculation loops in operation, and the SLO nomenclature is used for one recirculation loop in operation.

2.1. Major Contributors to SLMCPR Change

In general, the calculated safety limit is dominated by two key parameters: (1) flatness of the core bundle-by-bundle MCPR distribution, and (2) flatness of the bundle pin-by-pin power/R-Factor distribution. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR. MIP (MCPR Importance Parameter) measures the core bundle-by-bundle MCPR distribution and RIP (R-Factor Importance Parameter) measures the bundle pin-by-pin power/R-Factor distribution. The impact of the fuel loading pattern on the calculated TLO SLMCPR using rated core power and rated core flow conditions has been correlated to the parameter MIPRIP, which combines the MIP and RIP values.

Table 3 presents the MIP and RIP parameters for Cycle 11 and Cycle 12 along with the TLO SLMCPR estimate using the MIPRIP correlation. If the minimum core flow case is applicable, the TLO SLMCPR estimate is also provided for that case although the MIPRIP correlation is only applicable to the rated core flow case. This is done only to provide some reasonable

assessment basis of the minimum core flow case trend. In addition, Table 3 presents estimated impacts on the TLO SLMCPR due to methodology deviations, penalties, and/or uncertainty deviations from approved values. Based on the MIPRIP correlation and any impacts due to deviations from approved values, a final estimated TLO SLMCPR is determined. Table 3 also provides the actual calculated Monte Carlo SLMCPRs. Given the bias and uncertainty in the MIPRIP correlation [[ ]] and the inherent variation in the Monte Carlo results [[ ]], the change in the Limerick 2 Cycle 12 calculated Monte Carlo TLO SLMCPR using rated core power and rated core flow conditions is consistent with the corresponding estimated TLO SLMCPR value.

2.2. Deviations in NRC-Approved Uncertainties

Tables 4 and 5 provide a list of NRC-approved uncertainties along with values actually used. A discussion of deviations from these NRC-approved values follows; all of which are conservative relative to NRC-approved values. Also, estimated impact on the SLMCPR is provided in Table 3 for each deviation.

2.2.1. R-Factor

At this time, GNF has generically increased the GEXL R-Factor uncertainty from [[ ]] to account for an increase in channel bow due to the emerging unforeseen phenomena called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of the approved R-Factor uncertainty. The step "σ RPEAK" in Figure 4.1 from NEDC-32601P-A, which has been provided for convenience in Figure 3 of this attachment, is affected by this deviation. Reference 4 technically justifies that a GEXL R-Factor uncertainty of [[ ]] accounts for a channel bow uncertainty of up to [[ ]].

Limerick 2 has experienced control blade shadow corrosion-induced channel bow to the extent that an increase in the NRC-approved R-Factor uncertainty [[ ]] is deemed prudent to address its impact. Accounting for the control blade shadow corrosion-induced channel bow, the Limerick 2 Cycle 12 analysis shows an expected channel bow uncertainty of [[ ]], which is bounded by a GEXL R-Factor uncertainty of [[ ]]. Thus the use of a GEXL R-Factor uncertainty of [[ ]] adequately accounts for the expected control blade shadow corrosion-induced channel bow for Limerick 2 Cycle 12.

2.2.2. Core Flow Rate and Random Effective TIP Reading

In Reference 5 GNF committed to the expansion of the state points used in the determination of the SLMCPR. Consistent with the Reference 5 commitments, GNF performs analyses at the rated core power and minimum licensed core flow point in addition to analyses at the rated core power and rated core flow point. The approved SLMCPR methodology is applied at each state

point that is analyzed.

For the TLO calculations performed at 82.9% core flow, the approved uncertainty values for the core flow rate (2.5%) and the random effective TIP reading (1.2%) are conservatively adjusted by dividing them by 82.9/100. The steps "σ CORE FLOW" and "σ TIP (INSTRUMENT)" in Figure 4.1 from NEDC-32601P-A, which has been provided for convenience in Figure 3 of this attachment, are affected by this deviation, respectively.

Historically, these values have been construed to be somewhat dependent on the core flow conditions as demonstrated by the fact that higher values have always been used when performing SLO calculations. It is for this reason that GNF determined that it is appropriate to consider an increase in these two uncertainties when the core flow is reduced. The amount of increase is determined in a conservative way. For both parameters it is assumed that the absolute uncertainty remains the same as the flow is decreased so that the percentage uncertainty increases inversely proportional to the change in core flow. This is conservative relative to the core flow uncertainty since the variability in the absolute flow is expected to decrease somewhat as the flow decreases. For the random effective TIP uncertainty, there is no reason to believe that the percentage uncertainty should increase as the core flow decreases for TLO. Nevertheless, this uncertainty is also increased as is done in the more extreme case for SLO primarily to preserve the historical precedent established by the SLO evaluation. Note that the TLO condition is different than the SLO condition because for TLO there is no expected tilting of the core radial power shape.

The treatment of the core flow and random effective TIP reading uncertainties is based on the assumption that the signal to noise ratio deteriorates as core flow is reduced. GNF believes this is conservative and may in the future provide justification that the original uncertainties (non-flow dependent) are adequately bounding.

The core flow and random TIP reading uncertainties used in the SLO minimum core flow SLMCPR analysis remain the same as in the rated core flow SLO SLMCPR analysis because these uncertainties (which are substantially larger than used in the TLO analysis) already account for the effects of operating at reduced core flow.

2.2.3. LPRM Update Interval and Calculated Bundle Power

To adequately address the LPRM update/calibration interval in the Limerick 2 Technical Specifications, GNF has increased the LPRM update uncertainty in the SLMCPR analysis for Limerick 2 Cycle 12. The approved uncertainty values for the contribution to bundle power uncertainty due to LPRM update [[ ]] and the resulting total uncertainty in calculated bundle power [[ ]] are conservatively increased. The steps "σ TIP (INSTRUMENT)" and "σ BUNDLE (MODEL)" in Figure 4.1 from NEDC-32601P-A, which has been provided for convenience in Figure 3 of this attachment, are affected by this deviation.

[[

]] The total bundle power uncertainty is a function of the LPRM update uncertainty as detailed in Section 3.3 of NEDC-32694P-A.

2.3. Departure from NRC-Approved Methodology

No departures from NRC-approved methodologies were used in the Limerick 2 Cycle 12 SLMCPR calculations.

2.4. Fuel Axial Power Shape Penalty

At this time, GNF has determined that higher uncertainties and non-conservative biases in the GEXL correlations for the various types of axial power shapes (i.e., inlet, cosine, outlet and double hump) could potentially exist relative to the NRC-approved methodology values, see References 3, 6, 7 and 8. The following table identifies, by marking with an "X", this potential for each GNF product line currently being offered:

]]

]]

Axial bundle power shapes corresponding to the limiting SLMCPR control blade patterns are determined using the PANACEA 3D core simulator. These axial power shapes are classified in accordance to the following table:

[[

]]

If the limiting bundles in the SLMCPR calculation exhibit an axial power shape identified by this table, GNF penalizes the GEXL critical power uncertainties to conservatively account for the impact of the axial power shape. Table 6 provides a list of the GEXL critical power uncertainties determined in accordance to the NRC-approved methodology contained in NEDE-24011-P-A along with values actually used.

For the limiting bundles, the fuel axial power shapes in the SLMCPR analysis were examined to determine the presence of axial power shapes identified in the above table. These power shapes were not found; therefore, no power shape penalties were applied to the calculated Limerick 2 Cycle 12 SLMCPR values.

2.5. Methodology Restrictions

The four restrictions identified on Page 3 of NRC's Safety Evaluation relating to the General Electric Licensing Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A (March 11, 1999) are addressed in References 1, 2, 3, and 9.

No new GNF fuel designs are being introduced in Limerick 2 Cycle 12; therefore, the NEDC-32505P-A statement "...if new fuel is introduced, GENE must confirm that the revised R-Factor method is still valid based on new test data" is not applicable.

2.6. Minimum Core Flow Condition

For Limerick 2 Cycle 12, the minimum core flow SLMCPR calculation performed at 82.9% core flow and rated core power condition was limiting as compared to the rated core flow and rated core power condition. At low core flows, the search spaces for the limiting rod pattern and the nominal rod pattern are essentially the same. Additionally, the condition that MIP [[ ]] establishes a reasonably bounding limiting rod pattern. Hence, the rod pattern used to calculate the SLMCPR at 100% rated power/82.9% rated flow reasonably assures that at least 99.9% of the fuel rods in the core would not be expected to experience

boiling transition during normal operation or anticipated operational occurrences during the operation of Limerick 2 Cycle 12. Consequently, the SLMCPR value calculated from the 82.9% core flow and rated core power condition limiting MCPR distribution reasonably bounds this mode of operation for Limerick 2 Cycle 12.

2.7. Limiting Control Rod Patterns

The limiting control rod patterns used to calculate the SLMCPR reasonably assures that at least 99.9% of the fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences during the operation of Limerick 2 Cycle 12.

2.8. Core Monitoring System

For Limerick 2 Cycle 12, the 3DMonicore system will be used as the core monitoring system.

2.9. Power/Flow Map

The utility has provided the Cycle 11 and 12 power/flow map(s) in a separate attachment.

2.10. Core Loading Diagram

Figures 1 and 2 provide the core-loading diagram for Cycle 12 and 11 respectively, which are the Reference Loading Pattern as defined by NEDE-24011-P-A. Table 1 provides a description of the core.

2.11. Figure References

Figure 3 is Figure 4.1 from NEDC-32601P-A. Figure 4 is Figure III.5-1 from NEDC-32601P-A. Figure 5 is based on Figure III.5-2 from NEDC-32601P-A and has been updated with GE14 and GNF2 data. It has been reviewed and approved by the NRC as supported by Reference 10.

2.12. Additional SLMCPR Licensing Conditions

For Limerick 2 Cycle 12, no additional SLMCPR licensing conditions are included in the analysis.

2.13. Summary

The requested changes to the Technical Specification SLMCPR values are 1.09 for TLO and 1.12 for SLO for Limerick 2 Cycle 12.

3.0 References

1. 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.

2. 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.

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

4. Letter, John F. Schardt (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to Mel B. Fields (NRC), "Shadow Corrosion Effects on SLMCPR Channel Bow Uncertainty", FLN-2004-030, November 10, 2004.

5. Letter, Jason S. Post (GENE) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to Chief, Information Management Branch, et al. (NRC), "Part 21 Final Report: Non-Conservative SLMCPR", MFN 04-108, September 29, 2004.

6. Letter, Glen A. Watford (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to Alan Wang (NRC), "NRC Technology Update – Proprietary Slides – July 31 – August 1, 2002", FLN-2002-015, October 31, 2002.

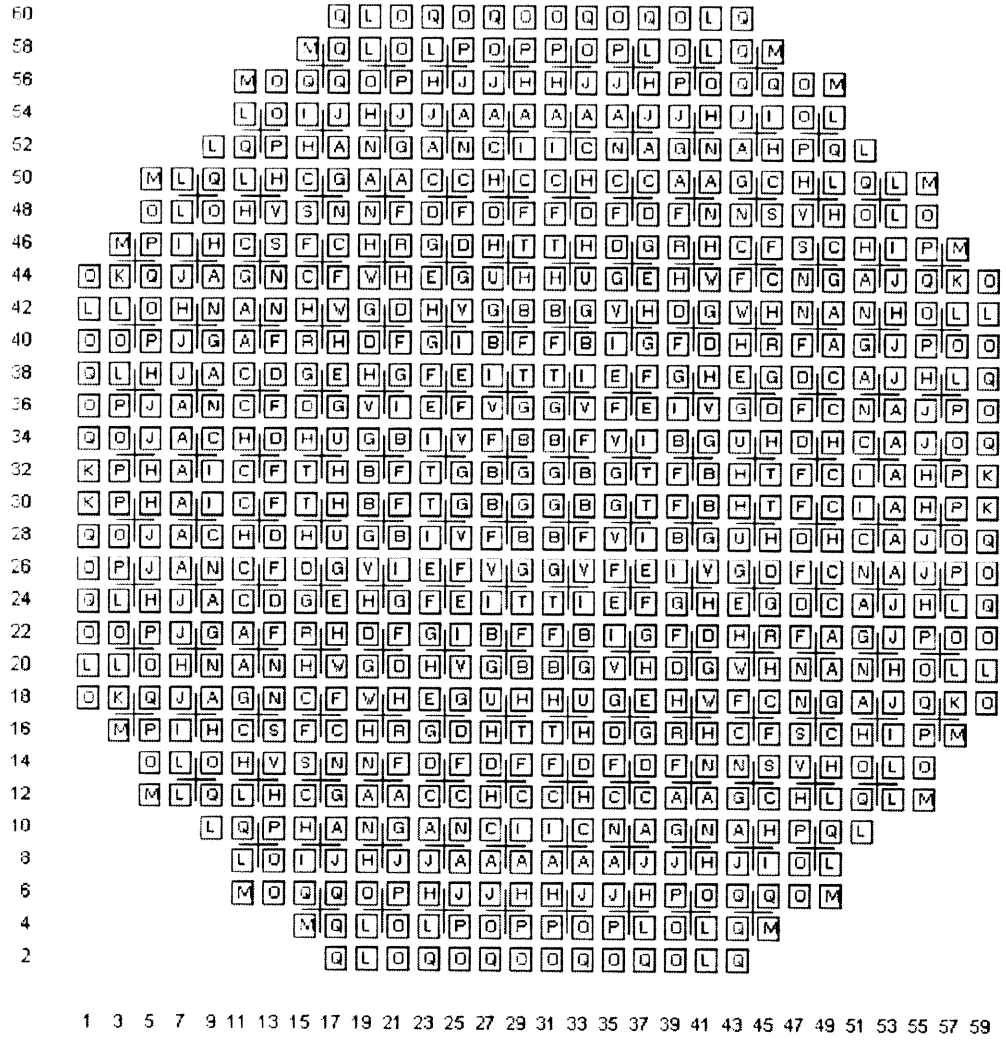
7. Letter, Jens G. Munthe Andersen (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to Alan Wang (NRC), "GEXL Correlation for 10X10 Fuel", FLN-2003-005, May 31, 2003.

8. Letter, Andrew A. Lingenfelter (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with cc to MC Honcharik (NRC), "Removal of Penalty Being Applied to GE14 Critical Power Correlation for Outlet Peaked Axial Power Shapes", FLN-2007-031, September 18, 2007.

9. Letter, Andrew A. Lingenfelter (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with cc to MC Honcharik (NRC), "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 2, June 2009 and GEXL Correlation for GNF2 Fuel, NEDC-33292P, Revision 3, June 2009", MFN 09-436, June 30, 2009.

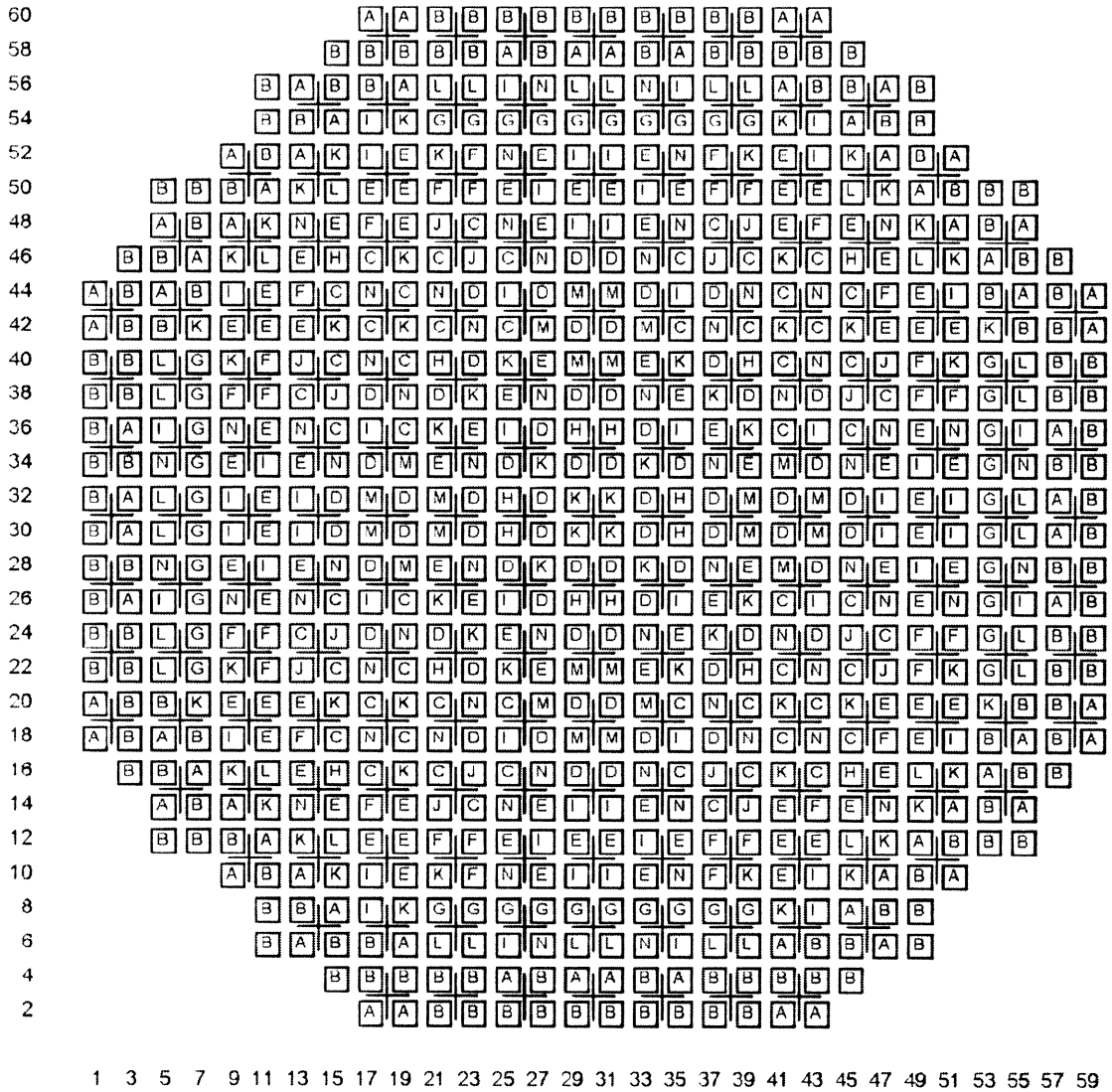
10. Letter, John D. Hughley (NRC) to Michael J. Pacilio (Exelon Generation Company, LLC), "Peach Bottom Atomic Power Station, Unit 2 – Issuance of Amendment Re: Safety Limit Minimum Critical Power Ratio Value Change (TAC No. ME3994)", ML102571768, September 28, 2010.

Figure 1. Cycle 12 Core Loading Diagram



Fuel Type	
A= GNF2-P10CG2B404-12G6.0-120T2-150-T6-3643	M= GE14-P10CNAB409-12GZ-120T-150-T6-2951
B= GNF2-P10CG2B386-2G8.0/13G7.0-120T2-150-T6-3644	N= GNF2-P10CG2B396-6G7.0/6G6.0-120T2-150-T6-3648
C= GNF2-P10CG2B389-14GZ-120T2-150-T6-3645	O= GE14-P10CNAB410-15GZ-120T-150-T6-2952
D= GNF2-P10CG2B391-15GZ-120T2-150-T6-3646	P= GE14-P10CNAB408-14GZ-120T-150-T6-2953
E= GNF2-P10CG2B392-14GZ-120T2-150-T6-3647	Q= GE14-P10CNAB410-13GZ-120T-150-T6-2955
F= GE14-P10CNAB389-14GZ-120T-150-T6-3157	R= GNF2-P10CG2B391-14GZ-120T2-150-T6-3725
G= GE14-P10CNAB389-15GZ-120T-150-T6-3158	S= GNF2-P10CG2B404-12G6.0-120T2-150-T6-3643
H= GE14-P10CNAB391-13GZ-120T-150-T6-3159	T= GNF2-P10CG2B389-14GZ-120T2-150-T6-3645
I= GE14-P10CNAB391-13G8.0-120T-150-T6-3160	U= GNF2-P10CG2B391-15GZ-120T2-150-T6-3646
J= GE14-P10CNAB390-12GZ-120T-150-T6-3161	V= GNF2-P10CG2B392-14GZ-120T2-150-T6-3647
K= GE14-P10CNAB409-16GZ-120T-150-T6-2956	W= GNF2-P10CG2B391-14GZ-120T2-150-T6-3725
L= GE14-P10CNAB410-16GZ-120T-150-T6-2950	

Figure 2. Cycle 11 Core Loading Diagram



Fuel Type	
A=GE14-P10CNAB400-15GZ-120T-150-T6-2832	H=GE14-P10CNAB409-16GZ-120T-150-T6-2956
B=GE14-P10CNAB397-14GZ-120T-150-T6-2833	I=GE14-P10CNAB410-16GZ-120T-150-T6-2950
C=GE14-P10CNAB389-14GZ-120T-150-T6-3157	J=GE14-P10CNAB409-12GZ-120T-150-T6-2951
D=GE14-P10CNAB389-15GZ-120T-150-T6-3158	K=GE14-P10CNAB410-15GZ-120T-150-T6-2952
E=GE14-P10CNAB391-13GZ-120T-150-T6-3159	L=GE14-P10CNAB408-14GZ-120T-150-T6-2953
F=GE14-P10CNAB391-13G8.0-120T-150-T6-3160	M=GE14-P10CNAB405-14GZ-120T-150-T6-2954
G=GE14-P10CNAB390-12GZ-120T-150-T6-3161	N=GE14-P10CNAB410-13GZ-120T-150-T6-2955

[[

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Figure 3. Figure 4.1 from NEDC-32601P-A

[[

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Figure 4. Figure III.5-1 from NEDC-32601P-A

[[

]]

Figure 5. Updated Figure III.5-2 from NEDC-32601P-A

Table 1. Description of Core

Description	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
Number of Bundles in the Core	764		764	
Limiting Cycle Exposure Point (i.e. BOC/MOC/EOC)	EOC	EOC	EOC	EOC
Cycle Exposure at Limiting Point (MWd/STU)	13000	13000	13350	13350
% Rated Core Flow	81.0	100.0	82.9	100.0
Reload Fuel Type	GE14		GNF2	
Latest Reload Batch Fraction, %	36.6		36.1	
Latest Reload Average Batch Weight % Enrichment	3.90		3.94	
Core Fuel Fraction: GE14 GNF2	1.000 0.000		0.639 0.361	
Core Average Weight % Enrichment	3.99		3.97	

Table 2. SLMCPR Calculation Methodologies

Description	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
Non-power Distribution Uncertainty	NEDC-32601P-A		NEDC-32601P-A	
Power Distribution Methodology	NEDC-32601P-A		NEDC-32601P-A	
Power Distribution Uncertainty	NEDC-32694P-A		NEDC-32694P-A	
Core Monitoring System	3DMonicore		3DMonicore	
R-Factor Calculation Methodology	NEDC-32505P-A		NEDC-32505P-A	

Table 3. Monte Carlo Calculated SLMCPR vs. Estimate

Description	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
[[				

Table 3. Monte Carlo Calculated SLMCPR vs. Estimate

Description	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case

]]

Table 4. Non-Power Distribution Uncertainties

	Nominal (NRC-Approved) Value ± σ (%)	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
GETAB					
Feedwater Flow Measurement	1.76	N/A	N/A	N/A	N/A
Feedwater Temperature Measurement	0.76	N/A	N/A	N/A	N/A
Reactor Pressure Measurement	0.50	N/A	N/A	N/A	N/A
Core Inlet Temperature Measurement	0.20	N/A	N/A	N/A	N/A
Total Core Flow Measurement	6.0 SLO/2.5 TLO	N/A	N/A	N/A	N/A
Channel Flow Area Variation	3.0	N/A	N/A	N/A	N/A
Friction Factor Multiplier	10.0	N/A	N/A	N/A	N/A
Channel Friction Factor Multiplier	5.0	N/A	N/A	N/A	N/A

Table 4. Non-Power Distribution Uncertainties

	Nominal (NRC-Approved) Value ± σ (%)	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
NEDC-32601P-A					
Feedwater Flow Measurement	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Feedwater Temperature Measurement	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Reactor Pressure Measurement	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Core Inlet Temperature Measurement	0.2	0.2	0.2	0.2	0.2
Total Core Flow Measurement	6.0 SLO/2.5 TLO	6.0 SLO/3.09 TLO	6.0 SLO/2.5 TLO	6.0 SLO/3.02 TLO	6.0 SLO/2.5 TLO
Channel Flow Area Variation	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Friction Factor Multiplier	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Channel Friction Factor Multiplier	5.0	5.0	5.0	5.0	5.0

Table 5. Power Distribution Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
GETAB/NEDC-32601P-A					
GEXL R-Factor	[[ ]]	N/A	N/A	N/A	N/A
Random Effective TIP Reading	2.85 SLO/1.2 TLO	N/A	N/A	N/A	N/A
Systematic Effective TIP Reading	8.6	N/A	N/A	N/A	N/A
NEDC-32694P-A, 3DMONICORE					
GEXL R-Factor	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Random Effective TIP Reading	2.85 SLO/1.2 TLO	2.85 SLO/1.48 TLO	2.85 SLO/1.2 TLO	2.85 SLO/1.45 TLO	2.85 SLO/1.2 TLO
TIP Integral	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Four Bundle Power Distribution Surrounding TIP Location	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Contribution to Bundle Power Uncertainty Due to LPRM Update	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]

Table 5. Power Distribution Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case
Contribution to Bundle Power Due to Failed TIP	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Contribution to Bundle Power Due to Failed LPRM	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Total Uncertainty in Calculated Bundle Power	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Uncertainty of TIP Signal Nodal Uncertainty	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]

Table 6. Critical Power Uncertainties

[[

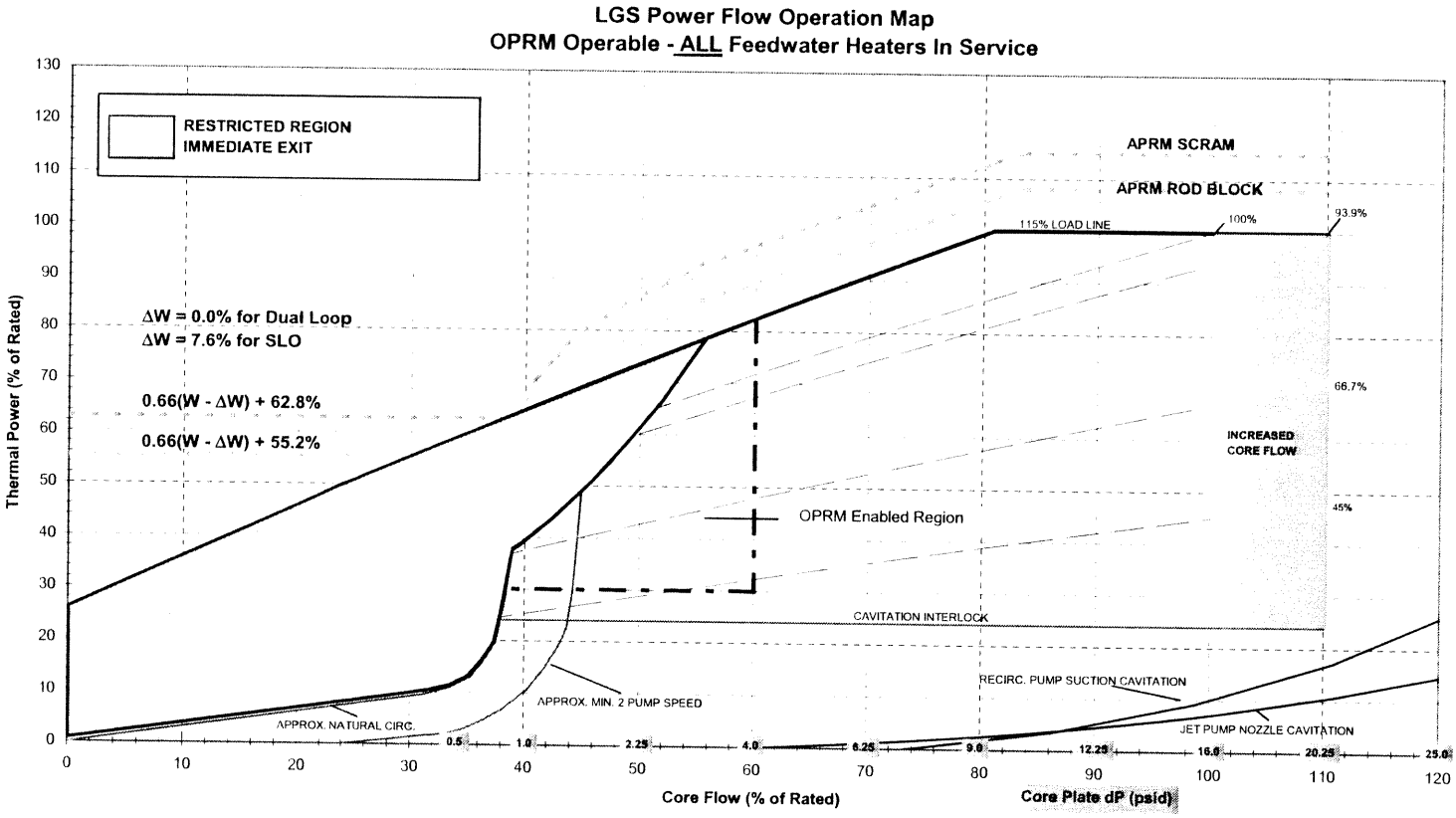
Description	Nominal Value ± σ (%)	Cycle 11 Minimum Core Flow Limiting Case	Cycle 11 Rated Core Flow Limiting Case	Cycle 12 Minimum Core Flow Limiting Case	Cycle 12 Rated Core Flow Limiting Case

]]

ATTACHMENT 6

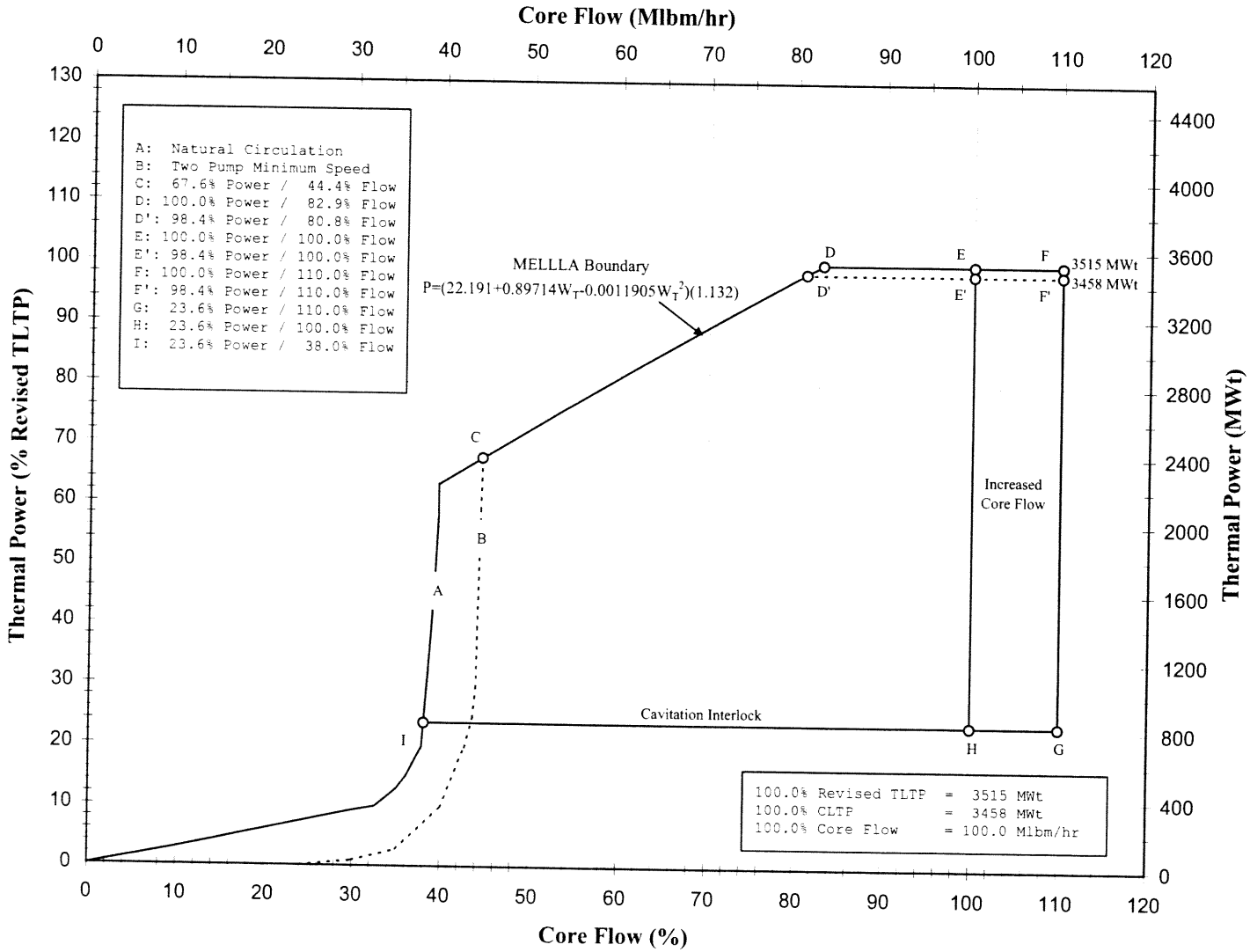
Power/Flow Maps for Cycles 11 and 12

Limerick Unit 2 Cycle 11



Limerick Unit 2 Cycle 12 (Draft)

Revised Limerick Power/Flow Map (Revised TPO – ~101.65% CLTP)



A final power/flow map is under development for Cycle 12. This is a draft.