Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon Nuclear

10 CFR 50.90

RS-02-214

December 26, 2002

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to Request for Additional Information Addressing Request for Amendment to Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," for Addition of New Analytical Methodology

Reference: (1) Letter from K. R. Jury (EGC) to U. S. NRC, "Request for Amendment to Technical Specification 5.6.5, 'Core Operating Limits Report (COLR),' for Addition of New Analytical Methodology," dated September 19, 2002.

The Reference 1 submittal transmitted a request by Exelon Generation Company (EGC), LLC, to revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18. Specifically, the proposed change added a new analytical method to TS Section 5.6.5, "Core Operating Limits Report (COLR)." During a discussion with the NRC on December 9, 2002, additional information was requested to support their review of Reference 1. Attached is the requested information.

If you have any questions or require additional information, please contact Mr. T. W. Simpkin at (630) 657-2821.

Sincerely,

T. W. Simpkin
for Keith R. Jury
Director – Licensing
Mid-West Regional Operating Group

Attachments: Affidavit
Response to Request for Additional Information

cc: Regional Administrator – NRC Region III
NRC Project Manager, NRR – LaSalle County Station
NRC Senior Resident Inspector – LaSalle County Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

A001

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF:	)	
EXELON GENERATION COMPANY (EGC), LLC	)	**Docket Numbers**
LASALLE COUNTY STATION - UNIT 1 and UNIT 2	)	**50-373 and 50-374**

SUBJECT: **Response to Request for Additional Information Addressing Request for Amendment to Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," for Addition of New Analytical Methodology**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information, and belief.

T. W. Simpkin
T. W. Simpkin
Manager-Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 26th day of

December, 2002

Anese L. Grigsby
Notary Public

OFFICIAL SEAL
ANESE L. GRIGSBY
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 3-13-2005

ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Question 1.

What types of fuel assemblies are loaded in the previous and current cycles?

Response 1.

LaSalle County Station (LSCS) utilized the following fuel inventory in its reactors. The current cycle for Unit 1 is cycle 10 and for Unit 2 is cycle 9.

	Unit 1	Unit 2
Cycle 8		ATRIUM-9B GE-9
Cycle 9	ATRIUM-9B GE-9	ATRIUM-9B GE-9
Cycle 10	ATRIUM-10 ATRIUM-9B GE-9	

Question 2.

Does LSCS update its supporting references? Does LSCS verify that all the approved methodologies listed in Technical Specification (TS) 5.6.5.b are still valid for its operation?

Response 2.

Yes, the supporting references utilized in the Core Operating Limits Report (COLR) are updated each cycle. The methodologies that are provided in TS 5.6.5.b are verified to be valid for operation of each cycle. The methodologies are reviewed against the vendor's calculation plan to ensure that the current NRC approved methodologies are utilized. If it is found that a methodology is referenced in the calculation plan that is not in the TS 5.6.5.b list but has received NRC approval, as was in the case for CASMO-4/MICROBURN-B2, a TS change is prepared to incorporate the methodology.

Question 3.

Which cycle-specific parameters listed in TS 5.6.5.a will be supported by CASMO-4/MICROBURN-B2?

Response 3.

CASMO-4/MICROBURN-B2 will be used to support the following cycle-specific parameters.

a) Average Planar Linear Heat Generation Rate (APLHGR)
b) Minimum Critical Power Ratio (MCPR)
c) Linear Heat Generation Rate (LHGR)
d) Rod Block Upscale Instrumentation Setpoint

ATTACHMENT
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Question 4.

Identify which parameter is supported by the approved methodologies listed in TS 5.6.5.b.

Response 4.

The following table cross references the methodologies listed in TS 5.6.5.b with the applicable TS LCO and justification.

TS 5.6.5b Report	Applicable TS LCO	Justification
1. ANF-1125 (P)(A), "ANFB Critical Power Correlation"	3.2.2	Presents an approved critical power correlation for ATRIUM-9 fuel
2. Letter, Ashok C. Thadani (NRC) to R. A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW spacer on 9X9-IX/X BWR Design", July 28, 1993	3.2.2	Input data for critical power correlation for 9X9-IX/X fuel design
3. XN-NF-524 (P)(A), "ANF Critical Power Methodology for Boiling Water Reactors"	3.2.2	Provides a methodology for the determination of thermal margins, specifically, the MCPR safety limit
4. ANF-913(P)(A), "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis"	3.2.2	Provides a computer program for analyzing BWR system transients
5. ANF-CC-33(P)(A), "HUXY: A Generalized Multirod Heatup Code with 10CFR50, Appendix K Heatup Option"	3.2.1	Develops a planar heat transfer model that is used to calculate peak cladding temperatures as part of the evaluation model methodology

6. XN-NF-80-19(P)(A), "Advanced Nuclear Fuel Methodology for Boiling Water Reactors"	3.2.1, 3.2.2, 3.2.3, 3.3.2.1	Provides an evaluation model methodology for licensing analysis of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10CFR50.46 and Appendix K, also describes the BWR core analysis methodology and produces input for nuclear plant transients and provides an overall methodology for determining MCPR operating limit .
7. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel"	3.2.3	Describes the process used in develop linear heat generation rates for fuel designs
8. ANF-89-014(P)(A), "ANF Corporation Generic Mechanical Design for ANF Corporation 9x9-IX and 9x9-9X BWR Reload Fuel"	3.2.3	Provides input data for ANF 9x9-IX and 9x9-9X BWR reload fuel
9. EMF-CC-074(P)(A), "Volume 4 – BWR Stability Analysis: Assessment of STAIF with input from MICROBURN-B2"	Indirectly impacts the MCPR and LHGR for core instability events	Describes methodology for stability analysis with input from the MICROBURN-B2 reactor core simulator
10. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model"	3.2.1, 3.2.2, 3.2.3	Provides an analytical capability to predict BWR fuel thermal and mechanical conditions for normal core operations and to establish initial conditions for power ramping, non-LOCA and LOCA analysis
11. XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis"	3.2.2	Provides a capability to perform analysis of transient heat transfer behavior in BWR assemblies

12. ANF-91-048(P)(A), "ANF Corporation Methodology for Boiling Water Reactor EXEM BWR Evaluation Model"	3.2.1	Describes modifications to the jet pump model in the RELAX blowdown code that better predict jet pump performance
13. EMF-2209(P)(A), "SPCB Critical Power Correlation"	3.2.2	Presents an improved critical power correlation for use with the ATRIUM-10 fuel designs
14. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs"	3.2.3	Establishes a set of design criteria that assures BWR fuel will perform satisfactorily throughout its lifetime
15. NEDE-24011-P-A, " GE Standard Application for Nuclear Fuel"	3.2.1, 3.2.2, 3.2.3, 3.3.2.1	GE neutronic, thermal-hydraulic, transient analysis and LOCA analysis methodology
16. NFSR-0085, "Benchmark of BWR Nuclear Design Methods"	3.2.1, 3.2.2, 3.2.3, 3.3.2.1	Exelon Corporation neutronic design methodology (GE methods)
17. NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods"	3.2.1, 3.2.2, 3.2.3, 3.3.2.1	Exelon Corporation neutronic design methodology benchmark (FANP methods)
18. EMF-1125(P)(A), "ANFB Critical Power Correlation Application for Co-resident Fuel"	3.2.2	Provide a critical power correlation for application with co-resident fuel
19. ANF-1125(P)(A), "ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties"	3.2.2	Provides the methodology to determine critical power correlation additive constant uncertainties
20. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model"	3.2.3	Extends the exposure limit of the RODEX2A code which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs