UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 5, 2009

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: CORE OPERATING LIMITS REPORT AND SCRAM TIME TESTING (TAC NO. MD9247)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 211 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 16, 2008, as supplemented by letters dated January 2 and March 19, 2009.

The amendment revises TSs 3.1.4, "Control Rod Scram Times," 3.2.2, "Minimum Critical Power Ratio (MCPR)," and 5.6.3, "Core Operating Limits Report (COLR)," to allow incorporation of the analytical methodologies associated with the operation of Global Nuclear Fuel-Americas (GNF) fuel into the licensing basis to support transition to GNF GE14 fuel.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 211 to NPF-21
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 211
License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Energy Northwest (licensee), dated July 16, 2008, as supplemented by letters dated January 2 and March 19, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-21 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 211 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to beginning operating cycle 20.

FOR THE NUCLEAR REGULATORY COMMISSION

Balwant K Singal for
Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. NPF-21
and Technical Specifications

Date of Issuance: May 5, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 211

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Facility Operating License No. NPF-21 and Appendix A, Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specification

REMOVE	INSERT
3.1.4-1	3.1.4-1
3.1.4-2	3.1.4-2
3.1.4-3	3.1.4-3
3.1.4-4	3.1.4-4
3.2.2-1	3.2.2-1
---	3.2.2-2
5.6-4	5.6-4

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

(6) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to store byproduct, source and special nuclear materials not intended for use at Columbia Generating Station. The materials shall be no more than 9 sealed neutron radiation sources designed for insertion into pressurized water reactors and no more than 40 sealed beta radiation sources designed for use in area radiation monitors. The total inventory shall not exceed 24 microcuries of strontium-90, 20 microcuries of uranium-235, 30 curies of plutonium-238, and 3 curies of americium-241.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3486 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 211 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

a. For Surveillance Requirements (SRs) not previously performed by existing SRs or other plant tests, the requirement will be considered met on the implementation date and the next required test will be at the interval specified in the Technical Specifications as revised in Amendment No. 149.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1, and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure $\geq$ 800 psig.	Prior to exceeding 40% RTP after each reactor shutdown $\geq$ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure $\geq$ 800 psig.	200 days cumulative operation in MODE 1
SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell AND Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1
Control Rod Scram Times

------------------------------------NOTES------------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 5. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds) WHEN REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
45	0.528
39	0.866
25	1.917
5	3.437

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig, are within established limits.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after $\geq$ 25% RTP <u>AND</u> 24 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

16. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation

17. EMF-CC-074(P)(A) Volume 4, "BWR Stability Analysis-Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation

18. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB Combustion Engineering Nuclear Operations

19. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications"

20. NEDC-33419P, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Global Nuclear Fuel

21. NEDE-24011-P-A and NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel (GESTAR II) and Supplement for United States," Global Nuclear Fuel

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 211 TO

FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated July 16, 2008 (Reference 1; Agencywide Documents Access and Management System (ADAMS) Accession No. ML082250678), as supplemented by letters dated January 2 and March 19, 2009 (References 2 and 3; ADAMS Accession Nos. ML090230569 and ML091040762, respectively), Energy Northwest (licensee) requested changes to the Technical Specifications (TSs; Appendix A to Facility Operating License No. NPF-21) for the Columbia Generating Station (CGS). The requested change would revise TSs 3.1.4, "Control Rod Scram Times," 3.2.2, "Minimum Critical Power Ratio (MCPR)," and 5.6.3, "Core Operating Limits Report (COLR)," to allow incorporating the analytical methodologies associated with operation of Global Nuclear Fuel-Americas (GNF) fuel into the licensing basis to support transition to GNF GE14 fuel.

The supplemental letters dated January 2 and March 19, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 14, 2008 (73 FR 60729).

Specifically, the licensee proposes to change TS 3.1.4 to adopt the licensing basis for the GNF methodology as follows:

1) Simplify the limiting condition for operation (LCO) and associated CONDITIONS and REQUIRED ACTIONS in accordance with NUREG-1433, "Standard Technical Specifications [STS], General Electric Plants, BWR/4." These changes consist of:

    - replacing current LCO 3.1.4 statement discussing average scram times in two-by-two arrays with two requirements. as follows:

        a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1, and

Enclosure 2

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

- replacing the ACTIONS section with the following:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met	A.1 Be in Mode 3.	12 hours

2) Change the NOTE above Table 3.1.4-1, "Control Rod Scram Times," to:

- add Note 1 - OPERABLE control rods with scram times not within the limits of this Table are considered "slow," and

- designate existing information as Note 2.

3) Change scram time limits to reflect the GNF analysis supported BWR/5 Scram Time versus Notch Position values.

4) Correct a typographical error in note (a) to change "as" to "at".

The licensee proposes to revise the frequency of TS surveillance requirement (SR) 3.1.4.1 and SR 3.1.4.4 by incorporating the changes specified by NRC-approved TS Task Force (TSTF) traveler TSTF-222-A, Revision 1, "Control Rod Scram Times," which modifies the STS to clarify the frequency of performing control rod scram time testing subsequent to performance of an outage that involved the movement of fuel. The licensee proposes to revise TS Section 3.1.4 to remove the surveillance test requirement to scram time test all control rods after each refueling outage. Only those control rods that reside in core cells that were affected by the refueling outage will need to be scram time tested after a refueling outage prior to reaching 40 percent rated thermal power (RTP). To affect this change, the frequency statements in SR 3.1.4.1 and SR 3.1.4.4 will be revised.

The licensee proposes to add new SR 3.2.2.2 to require MCPR operating limits to be determined subsequent to scram time testing required by SRs 3.1.4.1, 3.1.4.2, and 3.1.4.4. This surveillance will ensure that the specific scram speed distribution remains consistent with the GNF transient analysis that credits the conservatism in the actual scram speed performance. This additional SR is consistent with the proposed change in licensing basis to the GNF methodology and is reflected in the STS.

TS 5.6.3.b lists the analytical methods used by the licensee to determine core operating limits. The licensee proposes to add the following references to reflect the approval of the GNF methodology:

20. NEDC-33419P, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Global Nuclear Fuel

21. NEDE-24011-P-A and NEDE-24011-P-A-US, “General Electric Standard Application for Reactor Fuel (GESTAR II) and Supplement for United States,” Global Nuclear Fuel

2.0 REGULATORY EVALUATION

The NRC staff applied the following regulatory requirements during its review of the licensee's application.

In Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR), the Commission established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TS are required, in part, to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS.

Section 50.34(b) of 10 CFR Part 50 requires, in part, that:

> Each application for an operating license shall include a final safety analysis report. The final safety analysis report shall include...the following: (4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective [of assessing ... the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.]

As part of the reload design process, the licensee (or its vendor), performs reload safety analyses with approved methodologies to ensure that the design cycle will continue to meet the applicable regulatory criteria. To confirm that the analyses remain acceptable, the licensee confirms that key results of the safety analyses, such as the critical power ratio (CPR), are conservative with respect to the current design cycle. If key safety analysis results are not acceptable, a re-analysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

CGS TS 5.6.3, "Core Operating Limits Report (COLR)," requires, in part, that "[t]he analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC...."

Section 1.2.1.1.1, "Power Generation Design Criteria," of the Final Safety Analysis Report for CGS states that "Plant design conforms to applicable codes and regulations as stipulated in Table 1.2-1," which includes, in part, 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

10 CFR Part 50, Appendix A, General Design Criterion (GDC)-10, “Reactor design,” requires that:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design

> limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC-12, "Suppression of reactor power oscillations," requires that:

> The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

GDC-15, "Reactor coolant system design," requires that:

> The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC-29, "Protection against anticipated operational occurrences," requires that:

> The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

GDC-35, "Emergency core cooling," requires, in part, that:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The NRC staff applied the regulations in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and Appendix K, "ECCS Evaluation Models," of 10 CFR Part 50, in conjunction with the GDC, in its review of the licensee's application. GDC-35 is incorporated by reference into 10 CFR 50.46, which makes GDC-35 directly applicable to this review.

The NRC staff also used the regulatory guidance of applicable sections of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

3.0 TECHNICAL EVALUATION

The licensee currently operates the CGS reactor with AREVA-supplied ATRIUM-10 fuel and Westinghouse-supplied SVEA-96 fuel. The licensee plans to remove SVEA-96 fuel and insert GE14 fuel in the reactor during the upcoming refueling outage and will begin using GNF's safety analysis methodologies during the subsequent operating cycle 20. As part of its application, the licensee submitted licensing topical report, NEDC-33419-P, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel." The report contains the methodology, correlation, and associated uncertainties developed for modeling the AREVA ATRIUM-10 fuel design. This correlation will be applied to the ATRIUM-10 fuel that will remain in the reactor.

The licensee submitted information to demonstrate that GE14 fuel and the legacy fuel are thermal-hydraulically compatible, and that the GNF GESTAR II analytical methods are applicable to CGS for use in licensing calculations. The licensee analyzed the affected licensing basis events based on the GNF analytical methods and showed compliance with the applicable regulations.

The NRC staff evaluated the proposed transition of the fuel and the analytical methods, as documented in Sections 3.1 through 3.5 below, and evaluated the proposed TS changes as documented in Section 3.6 below.

3.1 GEXL97

The data for the GEXL97 development was generated using the NRC-approved SPCB correlation developed by AREVA (Reference 5; SPCB is the AREVA critical power correlation for ATRIUM-10 fuel). The database consisted of ATRIUM-10 sub-bundle and full bundle critical power data generated by the sub-channel code XCOBRA, incorporating the NRC-approved SPCB correlation. The objective of this data collection was to obtain ATRIUM-10 quality data appropriate for GEXL analysis.

The span of the data collection encompasses cosine, top-peaked, bottom-peaked, and double-humped axial power shapes. These data were generated to cover the complete range of expected operation of the ATRIUM-10 fuel in the CGS boiling-water reactor (BWR) core. The data were used to develop a new GEXL correlation for the ATRIUM-10 design, designated as GEXL97. The GEXL97 correlation uses the same functional form as previous GEXL correlations with different constants for the GEXL correlation coefficient parameters.

The GE critical quality-boiling length correlation (GEXL) was developed to predict the onset of boiling transition in BWR fuel assemblies during both steady-state and reactor transient conditions. The GEXL correlation is necessary for determining the MCPR operating limits resulting from transient analysis, the MCPR safety limit analysis, and the core operating performance and design. The GEXL correlation is an integral part of the transient analysis methodology. It is used to confirm the adequacy of the MCPR operating limit, and it can be used to determine the time of onset of boiling transition in the analysis of other events.

The NRC staff's review considered the following: 1) adequacy of the database generated with the sub-channel code XCOBRA, 2) proper determination of the uncertainty in GEXL97 correlation predictions for the ATRIUM-10 fuel design, 3) applicability of the proposed operating range of GEXL97 correlation to the ATRIUM-10 fuel.

3.1.1 Validity of the Database and Associated Uncertainties

ATRIUM-10 fuel is a 10x10 fuel bundle with a water channel design that displaces 9 fuel rods. It contains a total of 83 full-length fuel rods and 8 part-length rods. It has 27 unique fuel rod locations within the 10x10 lattice for which dryout data was collected.

The SPCB correlation for the ATRIUM-10 fuel, as encoded in the sub-channel computer code XCOBRA, is used to generate a database of predicted critical power values for a range of operating conditions corresponding to the range of the ATRIUM-10 correlation. This database was then treated as an empirical database, using the approved methodology for GEXL correlation development. Utilizing this approach, GNF produced a new form of the GEXL correlation, namely GEXL97, applicable only to the ATRIUM-10 fuel design.

The data for the GEXL97 development specific to ATRIUM-10 fuel was generated using the NRC-approved AREVA SPCB correlation encoded in the above stated sub-channel code. Specified rod-to-rod peaking factors, axial power shapes, pressure, mass flux and sub-cooling were used with the AREVA SPCB correlation to determine critical power at dryout.

Therefore, the NRC staff concludes that generating the analytical databases using the SPCB correlation encoded in the subchannel code XCOBRA is a reasonable engineering approach to dealing with mixed core fuel, where the experimental database and critical power correlation for the previous vendor's fuel is not available to the new vendor.

3.1.2 Determination of Uncertainties

The database used in the development of the GEXL97 correlation for ATRIUM-10 fuel was provided in Table 2-1 of NEDC-33419-P (Attachment 4 of Reference 1). This table shows the number of calculated critical power data points obtained using the AREVA critical power correlation for cosine, inlet, outlet, and double-humped axial power distributions. It also shows the fuel pin dryout location that formed the basis of the 28 different sets of AREVA calculated critical power data. Table 2-2 of the same document provides additional information by further dividing the data collected into subgroups of pressure, mass flux, and inlet sub-cooling.

The GEXL97 database generated in this manner is artificial in construct, created with a computer code which has implemented the SPCB correlation, and can at best only approximate the actual critical power raw data behavior of the ATRIUM-10 fuel. However, with reasonable engineering practices and proper statistical accountability, the database can predict the critical power behavior with acceptable uncertainties. Using the analytical database in the regression analysis introduces an additional uncertainty into the correlation being derived from it.

As stated earlier, the database for the GEXL97 development specific to ATRIUM-10 fuel was generated using the NRC-approved SPCB correlation. The database consisted of ATRIUM-10 sub-bundle and full bundle critical power data generated by the sub-channel code XCOBRA, incorporating the NRC-approved SPCB correlation.

The local critical power values predicted with the approved SPCB correlation can be expected to vary over the range of the database. Since the GEXL97 correlation is fitted to this analytical database, the error in the critical power prediction of the GEXL97 correlation for a given set of conditions will have some additional error relative to the real critical power value for those conditions, over and above the uncertainty of the correlation's fit to the analytical database. Therefore, the approach of the correlation procedure can be valid only if overall uncertainty in the new GEXL97 correlation is appropriately characterized in terms of the uncertainty in its fit to the analytical database and the uncertainty of the critical power values in the analytical database itself.

In the licensee's submittal, GNF appropriately combined the uncertainty of the fit of GEXL97 correlation to the analytical database and the uncertainty of the GEXL97 database, which is a function of the uncertainty of SPCB correlation. The NRC staff reviewed the treatment of the overall uncertainty of the GEXL97 correlation for ATRIUM-10 fuel, as presented in the licensee's submittal, and concluded that it was appropriate.

Therefore, the NRC staff concludes that the total uncertainty in the correlation's critical power predictions appropriately accounts for the uncertainty in the new correlation's fit to the analytical database and the uncertainty in the analytical database with respect to the underlying experimental data are appropriately treated.

3.1.3 Generation of the GEXL97 Correlation and the Range of Applicability

In developing the GEXL97, GNF took steps to optimize GEXL97 critical power predictions for the ATRIUM-10 fuel design, and to minimize the prediction uncertainty. This process is identical to that used by GNF when developing GEXL correlation coefficients for GNF fuel designs using raw test data, and has been used in past development of GEXL correlations applicable to other legacy fuel.

The procedure used for development of the GEXL97 correlation is summarized below:

a) First, a range of generated data covering all parameter variations is selected to form a correlation development database. This database consists of the majority of the generated data. A separate dataset is set aside to form a correlation verification database.

b) The GEXL97 correlation coefficients are then chosen to minimize the bias and standard deviation in correlating the development database, and to minimize any trend errors in reference to flow, pressure, subcooling, and R-factor (the R-factor is an input to the correlation that accounts for the effects of the fuel rod distributions and the fuel assembly and channel geometry on the fuel assembly critical power).

c) Once the optimum coefficients were determined, the apparent R-factors are calculated for each assembly. The apparent R-factor is defined as that R-factor which yields an overall experimental critical power ratio (ECPR) of 1.0 for a given assembly. ECPR is defined as the ratio of the GEXL97 calculated critical power to the SPCB calculated critical power.

d) A final set of additive constants (Table 4-2 of NEDC-33419-P) are determined by adjusting the preliminary additive constants subject to minimizing the difference between the R-Factors.

The range of application for the GEXL97 correlation, as stated in the submittal (Section 4.2 of NEDC-33419-P), is the same as the range of the analytical database over which the correlation is derived, and within the AREVA SPCB development database. The application range covers the complete range of expected operation of the ATRIUM-10 fuel during normal steady-state and transient conditions in the CGS BWR core. Therefore, the licensee's use of the new GEXL97 correlation within the limits of the analytical database, as bounded by the experimental limits derived from actual test data from the ATRIUM-10 database, is acceptable.

3.2 Thermal-Hydraulic Compatibility of the GE14 fuel with the ATRIUM-10 fuel

The supplemental information provided by the licensee (Reference 2) provided independent verification to the conclusion made by the fuel vendor that the GE14 and ATRIUM-10 fuels are thermal-hydraulically compatible. The next three cycles at CGS will be designated as mixed cores with the core comprised of ATRIUM-10 fuel and GE14 fuel. Cycle 20 will approximately consist of 2/3 core of ATRIUM-10 fuel and 1/3 core of GE14 fuel; Cycle 21 will approximately consist of 1/2 core of ATRIUM-10 fuel and 1/2 core of GE14 fuel; and Cycle 22 will approximately consist of 1/3 ATRIUM-10 fuel and 2/3 GE14 fuel. Consequently, GNF performed calculations to verify the mixed core calculations results regarding the similarity in thermal-hydraulic performance of the GE14 and ATRIUM-10 fuel designs. Data provided by AREVA was used by GNF to develop computer code models to perform the various evaluations.

Specifically, GNF investigated the thermal-hydraulic compatibility between GE14 and ATRIUM-10 through a series of mixed cores, progressing from the full core of ATRIUM-10 fuel to a full core of GE14 fuel. The mixed core analyses projected the performance of both fuel types during transition cores, from a full core of ATRIUM-10 fuel to a full core of GE14 fuel. During the core transition cycles, used (at least once-burned) ATRIUM-10 assemblies are placed at the core periphery.

GNF also performed evaluations to demonstrate compliance with safety and performance criteria, including core nuclear design and the thermal-hydraulic critical power correlations for the ATRIUM-10 fuel. GNF calculations provided confirmation that the thermal-hydraulic performance characteristics applied in the calculations met specific acceptance criterion associated with the thermal-hydraulic compatibility of GE14 fuel and the legacy fuel.

Based on the above, the NRC staff concluded that the GE14 fuel was thermal-hydraulically compatible with the legacy ATRIUM-10 fuel.

3.3 Use of Approved Analytical Methods

Analytical methods (e.g., computer codes, correlations, etc.) used to support licensing calculations are generally documented in topical reports (TRs), which may be reviewed by the NRC staff on a generic (i.e., stand-alone) basis. In an NRC staff safety evaluation (SE) approving a typical TR, the staff defines the basis for acceptance in conjunction with any limitations and conditions on use of the TR, as appropriate. Therefore, the NRC staff requested that the licensee document that the use of the proposed analytical methods for CGS is consistent with the staff approval.

In Reference 2, the licensee listed the TRs and analytical methods used for each affected analysis and stated that such use was consistent with the corresponding staff approval. Therefore, the NRC staff concludes that the analyses used for the new fuel are acceptable. The analytical methods used for CGS are summarized below in Section 3.4.4, Table 1.

3.4 Licensing Basis Analyses

In Reference 3, the licensee provided plant-specific information to support the methodology change in conjunction with the introduction of GE14 fuel. Specifically, the licensee performed analyses of the limiting final safety analysis report (FSAR) events with the GNF methods to demonstrate that the results of the analysis meet the applicable acceptance criteria. The information provided by the licensee is summarized below in Section 3.4.4, Table 1.

The events analyzed include:

- Limiting Anticipated Operational Occurrences (AOOs; Turbine Trip with no Bypass, Load Rejection with no Bypass, Feedwater Controller Failure)
- American Society of Mechanical Engineers (ASME) Overpressure (Main Steam Isolation Valve Closure with Flux Scram)
- Stability
- Emergency Core Cooling System (ECCS) – Loss-of-Coolant Accident (LOCA)
- Anticipated Transient without Scram (ATWS; Main Steam Isolation Valve Closure, Pressure Regulator Failure Open)

3.4.1 AOO and ASME Overpressure

The plant responses to the limiting AOOs are analyzed by the licensee for each reload cycle to establish the operating limit minimum critical power ratio (OLMCPR). The ASME overpressure analysis is also performed every cycle to ensure that vessel pressurization following a limiting transient is within the acceptable limit. The licensee performed the reload transient analysis to cover the projected operating conditions within the licensed power-to-flow map, the expected core exposures, and equipment availability conditions. In Reference 3, the licensee provided the OLMCPR limits for the Cycle 20 core containing GE14 and co-resident ATRIUM-10 fuel. In addition, the licensee showed that ASME overpressure results are acceptable. The results of the analysis, as provided by the licensee in Reference 3, satisfy the design requirements of GDC-10, regarding the minimum critical power ratio (MCPR), which protects fuel integrity during normal operations and AOOs, and GDC-15, regarding RCS pressure boundary integrity for normal operations and AOOs. Therefore, the NRC staff concludes that the analyses used for the new fuel are acceptable.

3.4.2 Stability

CGS is currently operating under the requirements of the reactor stability Long-Term Solution Option III, approved by the NRC staff in TRs NEDO-32465-A and NEDO-31960-A (References 9 and 10). Option III is a solution based on detection and suppression of instabilities. The figure of merit is the MCPR. The stability-based operating limit MCPR (OLMCPR) values are calculated as a function of Oscillation Power Range Monitor (OPRM) amplitude setpoints, ranging from 1.05 to 1.15. The stability-based OLMCPR values are calculated for two postulated instability events: steady-state operation OLMCPR(SS) and dual recirculation pump trip OLMCPR(2PT). The OPRM amplitude setpoint is chosen so that the OLMCPR remains greater than or equal to the stability-based OLMCPR, thereby ensuring safety limit MCPR protection.

In addition to the detection and suppression of instabilities, backup stability protection (BSP) regions are calculated to exclude or limit operation in regions on the power/flow map susceptible to thermal-hydraulic instabilities. The licensee validated the BSP regions based on the approved ODYSY methodology (References 11, 12, and 13).

The results of the analysis demonstrate that (1) the new fuel will not exceed the acceptable fuel design limits, specifically MCPR, as required by GDC-10, and (2) power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be readily detected and suppressed, as required by GDC-12. Therefore, the NRC staff concludes that the analyses used for the new fuel are acceptable.

3.4.3 ECCS-LOCA

The ECCS is designed to mitigate postulated LOCAs due to ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46, which references 10 CFR Part 50, Appendix K. The analysis methodology used by the licensee for CGS LOCA analysis is the SAFER/GESTR-LOCA evaluation model (References 14 through 18), which has been approved by the NRC.

To support the transition, the licensee performed the LOCA analysis for the GE14 core. The licensee examined break sizes ranging from the double-ended guillotine break of recirculation suction line to small breaks for which no core heat-up is predicted. The limiting large break was identified as the double-ended guillotine break of the recirculation suction line and the limiting small break was identified as a 0.07 ft^2 break in the recirculation suction line. The 0.07 ft^2 break was identified as the overall most limiting break assuming the worst single failure of high pressure core spray diesel generator. The licensee evaluated both mid-peaked and top-peaked axial power shapes. The limiting axial power shape was identified as mid-peaked for large breaks and top-peaked for small breaks.

The licensee evaluated potentially limiting power and flow conditions including the Extended Loadline Limit Analysis and increased core flow conditions. For this analysis, the licensee identified the rated power/flow point as the limiting power to flow condition. The licensee also concluded that the single-loop operation is bounded by the two-loop operation, and reduced feedwater temperature is bounded by normal feedwater temperature.

Thermal-hydraulic compatibility is demonstrated for the legacy fuel and the GE14 fuel; consequently, the mixed core does not invalidate the legacy fuel maximum average planar linear heat generation rates.

The NRC staff finds that the licensee provided sufficient information to demonstrate compliance with the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

3.4.4 ATWS

ATWS is defined as an AOO followed by the failure of the reactor protection system. To demonstrate acceptability in accordance with the CGS final safety analysis report, the ATWS analysis must show that 1) the peak vessel bottom pressure is less than the ASME service level C limit of 1500 pounds per square inch gauge (psig); 2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 degrees Fahrenheit (°F); 3) the peak suppression pool temperature is less than the design limit (204.5 °F for CGS); and 4) the peak containment pressure is less than the containment design pressure (45 psig for CGS). The licensee performed the ATWS analysis for CGS Cycle 20 conditions and showed that the analysis results meet the acceptance criteria, as shown in Table 1.

Based on the information submitted by the licensee, the NRC staff finds that the results of the analysis meet the applicable acceptance criteria. The licensee confirmed that the CPR safety analyses remain bounding, and that key inputs to the safety analyses are conservative with respect to the current design cycle. Therefore, the staff concludes that the impact of the fuel and methodology change on the safety analysis for CGS is acceptable.

Table 1 - Limiting Analysis Results and Computer Codes and Methodology Used

Analysis	Code(s) Used	Staff Approval	Key Parameter(s)	Result vs. (Acceptance Criteria)
AOO	ISCOR09 PANAC11 ODYN09V TASC03	References 6, 7, and 8	MCPR	Base case OLMCPR range of 1.30 to 1.39 (See Note 1)
ASME Overpressure	ISCOR09 PANAC11 ODYN09V	References 6 and 7	Peak Dome Pressure (psig) Peak Vessel Pressure (psig)	1305 (≤1325) 1341 (≤1375)
Stability	ISCOR09 PANAC11 ODYSY05	References 9, 10, 11, 12, and 13	OLMCPR BSP regions	OLMCPR(SS) range of 1.244 to 1.514, OLMCPR (2PT) range of 1.184 to 1.440 (See Note 2)
ECCS-LOCA	ISCOR09 LAMB08 SAFER04/ GESTR08 TASC03	References 14, 15, 16, 17, and 18	PCT (°F) Max local oxidation (%) Core-wide Metal-water reaction (%)	1710 (≤2200) 1 (≤17) 0.1 (≤1.0)
ATWS	ISCOR09 PANAC11 ODYN09V STEMP04 TASC03	References 6, 7, and 8	Peak Vessel Pressure (psig) Peak Suppression Pool Temperature (°F) Peak Containment Pressure (psig) Peak Cladding Temperature (°F) Peak Local Cladding Oxidation (%)	1449 (≤1500) 189.4 (≤204.5) 12.4 (≤45) 1542 (≤2200) Insignificant (≤17) (See Note 3)

Note 1: The AOO analysis determines the OLMCPR such that the limiting transient would not violate the safety limit MCPR. In Reference 3, the licensee provided a base case Option B OLMCPR of 1.30 for beginning of cycle (BOC) to middle of cycle (MOC), and 1.39 for MOC to end of cycle (EOC), with all equipment operable.

Note 2: In Reference 3, the licensee provided the stability-based OLMCPR values for the two scenarios: 1) steady-state operation, OLMCPR(SS) and 2) dual recirculation pump trip, OLMCPR(2PT). The OPRM amplitude setpoint is selected by the licensee such that the limiting stability event would not violate the safety limit MCPR. The BSP regions were calculated for nominal and reduced feedwater temperature conditions based on core and channel decay ratio criteria.

Note 3: The calculated peak cladding temperature is less than 1600 °F, and therefore cladding oxidation is insignificant compared to the acceptance criteria and is not explicitly calculated.

3.5 Fuel and Methodology Change – Conclusions

In consideration of information discussed in Sections 3.1 through 3.4, the NRC staff finds that the proposed fuel and methodology transition is acceptable. The staff concludes that the use of GEXL97 is acceptable for the following reasons:

a) the total uncertainty in the correlation's critical power predictions appropriately accounts for the uncertainty in the new correlation's fit to the analytical database and the uncertainty in the analytical database with respect to the underlying experimental data are appropriately treated;

b) generating the analytical databases using the SPCB correlation encoded in the sub-channel code XCOBRA is a reasonable engineering approach to dealing with mixed core fuel, where the experimental database and critical power correlation for the previous vendor's fuel is not available to the new vendor;

c) GNF intends to utilize the new GEXL97 correlation within the limits of the analytical database, bounded by the experimental limits of the ATRIUM-10 database; and

d) GNF confirmed that the CPR analyses remain bounding, and that key inputs to the safety analyses (such as the CPR) are conservative with respect to the current design cycle.

In addition, the staff finds that the introduction of the GE14 fuel will not adversely affect the performance of the ATRIUM-10 fuel, and that the two distinct fuel designs are thermal-hydraulically compatible. The staff finds that the use of the proposed analytical methods for CGS is consistent with the corresponding NRC staff approval and that the results of the analyses meet the applicable acceptance criteria. Therefore, the staff concludes that the proposed fuel and methodology transition for CGS is acceptable.

3.6 Technical Specification Changes

3.6.1 TS 5.6.3, "Core Operating Limits Report (COLR)"

TS 5.6.3.b provides a list of TRs documenting the NRC-approved methodologies used to determine the values of cycle-specific parameters included in the COLR. The license amendment request proposes to add the following TRs to the reference list:

20. NEDC-33419P, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Global Nuclear Fuel

21. NEDE-24011-P-A and NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel (GESTAR II) and Supplement for United States," Global Nuclear Fuel

As discussed above in Section 3.0, the staff finds that the GEXL97 method documented in the referenced TR is acceptable for use in support of CGS licensing applications. Therefore, the staff finds that the addition of the GEXL97 method to the COLR is acceptable.

As discussed above in Section 3.0, the licensee submitted information demonstrating the applicability of GESTAR II and associated safety analysis methodologies to CGS. Therefore, the staff finds that the addition of GESTAR II to the COLR is acceptable for referencing in CGS licensing applications.

3.6.2 TS LCO 3.1.4, "Control Rod Scram Times" and Table 3.1.4-1 "Control Rod Scram Times"

The proposed change revises the current TS LCO 3.1.4 and the associated action statement to adopt a different scram time testing method, which is used to determine if the measured scram insertion times are sufficient to provide sufficient negative reactivity assumed in the licensing basis analysis. The proposed change also revises the allowed scram time values in TS Table 3.1.4-1.

The current testing method used by the licensee places requirements on maximum individual control rod drive (CRD) insertion times (7.0 second requirement) and average scram insertion times for two-by-two arrays. The proposed change simplifies the testing method by basing the scram time acceptability on individual control rod performance. Basing the scram time acceptability on individual control rod performance also eliminates the concern of potentially allowing operation with too many "slow" rods because a few fast scramming rods were available to provide an acceptable "average time" of multiple rods. The revised LCO would allow no more than 13 control rods (or approximately 7 percent of 185) to be "slow" and allows no more than two "slow" control rods to occupy adjacent locations.

The purpose of TS LCO 3.1.4 is to ensure that the actual scram performance of the individual control rods supports the assumed negative scram reactivity in the licensing analysis. As the basis for the proposed TS LCO 3.1.4, the licensee referenced BWROG-8754, dated September 1987 (Reference 4). The analysis discussed in BWROG-8754 provides that the analytical scram reactivity curve will be satisfied if no more than 7 percent of the rods are slow and these slow rods are distributed in satisfactory manner. The NRC staff previously reviewed and accepted the basis used in BWROG-8754 and incorporated it in Revision 3 of the STS. The licensee proposes to simplify the corresponding LCO 3.1.4 action statement to match the STS. The licensee proposes to revise the allowed scram time values in Table 3.1.4-1 to values to be consistent with BWROG-8754. Use of the NRC-approved BWROG-8754 basis ensures that the actual scram performance of the individual control rods continues to support the assumed negative scram reactivity in the licensing analysis.

The NRC staff notes that, as part of the standard reload licensing process, the licensee must perform a cycle-specific safety analysis consistent with the TS requirements and demonstrate compliance with the licensing basis.

Based on above, the NRC staff finds the proposed changes to TS LCO 3.1.4 and Table 3.1.4-1 are acceptable.

3.6.3 TS SR 3.1.4.1 and TS SR 3.1.4.4

Current CGS TS SR 3.1.4.1 requires that the scram time of each control rod be verified to be within TS Table 3.1.4-1 acceptance criteria prior to exceeding 40 percent RTP after a refueling or after a shutdown of 120 days or greater. The licensee proposes to revise SR 3.1.4.4 to require scram time testing of the control rod prior to exceeding 40 percent RTP after fuel movement within the affected core cell. In a typical routine refueling outage, all core cells are likely to be affected as a result of some fuel movement (e.g., a spent fuel assembly is replaced with a fresh assembly, a fuel assembly is relocated from one cell to another, or a fuel assembly is reoriented within a core cell). Therefore, scram time testing will continue to be conducted on essentially all control rods following a routine refueling.

However, if a core cell is not affected by (1) movement of one of the four fuel assemblies in the cell, (2) replacement of the control rod in that cell, or (3) maintenance on the CRD system for the rod in that cell, the scram time of the control rod in that core cell is not expected to be impacted. As a result, there would be no need to conduct scram time testing on that unaffected control rod. Furthermore, it is expected that the periodic scram time testing of a representative sample (10 percent of the control rods), as required by SR 3.1.4.2, will identify any long-term phenomenon that could result in degradation of scram time. Revising the second frequency requirement of SR 3.1.4.4 to require scram time testing after fuel movement "within the affected core cells," clarifies that only those control rods in core cells in which fuel was moved or replaced or control rod maintenance was performed are required to be scram time tested. It is expected that all core cells will be affected in this manner during a routine refueling outage and, therefore, the scram time testing will be required on essentially all control rods.

Deleting the first frequency requirement of SR 3.1.4.1, revising the second frequency requirement of SR 3.1.4.4, and reversing the order of the two frequency requirements of SR 3.1.4.4 is consistent with the STS and NRC-approved TSTF-222-A, Revision 1. The proposed change clarifies that post-fueling control rod scram time testing only applies to control rods affected by movement of fuel. Therefore, the NRC staff finds the proposed changes to SR 3.1.4.1 and SR 3.1.4.4 acceptable.

3.6.3 TS SR 3.2.2.2

The licensee proposes to add new SR 3.2.2.2, a requirement to determine the MCPR operating limit after performance of various scram time surveillances. After a scram time test is performed via SRs 3.1.4.1, 3.1.4.2, or 3.1.4.4, the data from these tests is used to generate the actual scram speed distribution which is then compared with the assumed distribution used by the transient analysis. This comparison is then used as an input to determine the OLMCPR.

By performing this surveillance within 72 hours of obtaining the necessary actual scram time input data, the effective scram speed distribution can be monitored to ensure that it remains consistent with the transient analysis. This approach is consistent with the Revision 3 of the STS. When the licensee adopted STS in 1997, this SR was not adopted because the contracted fuel vendor at the time did not use this analytical approach. Adoption of the SR at this time is consistent with the use of GNF fuel and the GNF analytical methodology. Therefore, the NRC staff finds the proposed changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Washington State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on October 14, 2008 (73 FR 60729). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from Sudesh K. Gambhir, Energy Northwest, to U.S. Nuclear Regulatory Commission, NRC, "License Amendment Request for Changes to Technical

Specifications Involving Core Operating Limits Report and Scram Time Testing," dated July 16, 2008 (ADAMS Accession No. ML082250680).

2. Letter from Sudesh K. Gambhir, Energy Northwest, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding License Amendment Request Involving Core Operating Limits Report and Scram Time Testing," dated January 2, 2009 (ADAMS Accession No. ML090230569).

3. Letter from Sudesh K. Gambhir, Energy Northwest, to U.S. Nuclear Regulatory Commission, "Columbia Generating Station, Docket No. 50-397, Supplemental Response to Request for Additional Information (RAI) Regarding License Amendment Request Involving Core Operating Limits Report and Scram Time Testing," dated March 19, 2009 (ADAMS Accession No. ML091040762).

4. Letter from R. F. Janecek, Boiling Water Reactor Owners Group, to R.W. Starostecki, U.S. Nuclear Regulatory Commission, "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, dated September 17, 1987.

5. EMF-2209(P)(A) Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003.

6. NEDE-30130-P-A, "Steady-State Nuclear Methods," April 1985, and for PANACEA11, Letter from Stuart A. Richards, U.S. Nuclear Regulatory Commission, to Glen A. Watford, General Electric Nuclear Energy, "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods (TAC NO. MA6481)," dated November 10, 1999 (ADAMS Accession No. ML993230184)

7. NEDC-24154P-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors, Supplement 4, Volume 1," January 1998.

8. NEDC-32084P-A, "TASC-03A, A Computer Program for Transient Analysis of a Single Channel, Revision 2," July 2002.

9. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

10. NEDO-31960-A, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology, including Supplement 1," November 1995.

11. NEDC-32992P-A, "ODYSY Application for Stability Licensing Calculations," July 2001.

12. NEDE-33213P, "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," May 2007.

13. Letter from Sheri L. Bone, U.S. Nuclear Regulatory Commission, to Doug Coleman, Energy Northwest, "Final Safety Evaluation for Boiling Water Reactors Owners' Group (BWROG) Licensing Topical Report (LTR) NEDE-33213P, 'Application for Stability

Licensing Calculations Including Option I-D and II Long Term Solutions' (TAC No. MD5743)," dated February 23, 2009 (ADAMS Accession No. ML090090032).

14. NEDE-20566P-A, "General Electric Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K," September 1986.

15. NEDE-23785-1-P-A Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 1, GESTR-LOCA - A Model for the Prediction of Fuel Rod Thermal Performance," October 1984.

16. NEDE-23785-1-P-A Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 2, SAFER - Long Term Inventory Model for BWR Loss-of-Coolant Analysis," October 1984.

17. NEDE-23785-1-P-A Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 3, SAFER/GESTR Application Methodology," October 1984.

18. NEDE-23785P-A Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 3 Supplement 1, Additional Information for Upper Bound PCT Calculation," March 2002.

Principal Contributors: T. Nakanishi
A. Attard

Date: May 5, 2009

May 5, 2009

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: CORE OPERATING LIMITS REPORT AND SCRAM TIME TESTING (TAC NO. MD9247)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 211 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 16, 2008, as supplemented by letters dated January 2 and March 19, 2009.

The amendment revises TSs 3.1.4, "Control Rod Scram Times," 3.2.2, "Minimum Critical Power Ratio (MCPR)," and 5.6.3, "Core Operating Limits Report (COLR)," to allow incorporation of the analytical methodologies associated with the operation of Global Nuclear Fuel-Americas (GNF) fuel into the licensing basis to support transition to GNF GE14 fuel.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 211 to NPF-21
2. Safety Evaluation

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