Progress Energy

Michael J. Annacone
Vice President
Brunswick Nuclear Plant

APR 29 2010

SERIAL: BSEP 10-0057
TSC-2010-02

10 CFR 50.90

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324
Renewed Facility License Nos. DPR-71 and DPR-62
Request for License Amendments – Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)"

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., is requesting a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments revise Technical Specification (TS) 5.6.5.b by adding AREVA Topical Report BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008, to the list of analytical methods that have been reviewed and approved by the NRC for determining core operating limits. An evaluation of the proposed license amendments is provided in Enclosure 1.

CP&L has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and determined that this change involves no significant hazards considerations.

In accordance with 10 CFR 50.91(b), CP&L is providing a copy of the proposed license amendment to the designated representative for the State of North Carolina.

CP&L requests approval of the proposed amendments by March 4, 2011, in order to support reactor start-up following the Unit 2 refueling outage, which is currently scheduled to begin on March 5, 2011. Once approved, the Unit 2 amendment shall be implemented prior to start-up from the 2011 Unit 2 refueling outage and the Unit 1 amendment shall be implemented prior to start-up from the 2012 Unit 1 refueling outage.

No regulatory commitments are contained in this submittal. Please refer any questions regarding this submittal to Ms. Annette Pope, Supervisor – Licensing/Regulatory Programs, at (910) 457-2184.

A001
NRR

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on April 29, 2010.

Sincerely,

Michael J. Annacone

WRM/wrm

Enclosures:

1. Evaluation of License Amendment Request
2. Marked-up Technical Specification Pages - Unit 1
3. Typed Technical Specification Pages - Unit 1
4. Typed Technical Specification Pages - Unit 2

cc (with enclosures):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Luis A. Reyes, Regional Administrator
245 Peachtree Center Avenue, NE, Suite 1200
Atlanta, GA 30303-1257

U. S. Nuclear Regulatory Commission
ATTN: Mr. Philip B. O'Bryan, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission **(Electronic Copy Only)**
ATTN: Mrs. Farideh E. Saba (Mail Stop OWFN 8G9A)
11555 Rockville Pike
Rockville, MD 20852-2738

Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

Mr. W. Lee Cox, III, Section Chief
Radiation Protection Section
North Carolina Department of Environment and Natural Resources
1645 Mail Service Center
Raleigh, NC 27699-1645

Evaluation of Proposed License Amendment Request

Subject: Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)"

1.0 Description

This letter is a request by Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., to amend the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments revise Technical Specification (TS) 5.6.5.b by adding AREVA Topical Report BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008, to the list of analytical methods that have been reviewed and approved by the NRC for determining core operating limits.

2.0 Proposed Change

The proposed amendments will add AREVA Topical Report BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008, to the list of analytical methods specified in Technical Specification 5.6.5.b that have been reviewed and approved by the NRC for determining core operating limits.

For convenience, Enclosure 2 contains a marked-up version of the Unit 1 Technical Specifications showing the proposed changes. Since Technical Specification 5.6.5.b for Unit 1 and Unit 2 is identical, only the mark-up for Unit 1 is provided. Enclosures 3 and 4 provide typed versions of the Unit 1 and Unit 2 Technical Specifications, respectively. These typed Technical Specification pages are to be used for issuance of the proposed amendments.

3.0 Background

Core operating limits are established each operating cycle in accordance with Technical Specification 3.2, "Power Distribution Limits" and Technical Specification 5.6.5, "Core Operating Limits Report (COLR)." These operating limits ensure that the fuel design limits are not exceeded during any conditions of normal operation and in the event of any Anticipated Operational Occurrence (AOO). The methods used to determine the operating limits are those previously found acceptable by the NRC and listed in Technical Specification 5.6.5.b.

On February 12, 2008 (i.e., ADAMS Accession Number ML080350138), in response to AREVA's application dated August 19, 2004 (i.e., ADAMS Accession Number ML042810356), the NRC staff found that Topical Report BAW-10247(P), Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, is acceptable for referencing in licensing applications for boiling water reactors to the extent specified and under the limitations delineated in the topical report and the NRC's final safety evaluation. By letter dated May 2, 2008 (i.e., ADAMS Accession Number ML081340207), AREVA published the accepted version of the topical report, BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008.

On March 27, 2008 (i.e., ADAMS Accession Number ML080930637), in response to CP&L's application dated January 22, 2007 (i.e., ADAMS Accession Number ML070300570), the NRC issued License Amendments 246 and 274 for BSEP, Units 1 and 2, respectively, revising the Technical Specifications to support use of AREVA fuel and core design methodologies. Beginning with the Cycle 17 reactor core for BSEP, Unit 1 and Cycle 19 core for BSEP, Unit 2, CP&L began using AREVA fuel and core design methodologies to determine core operating limits.

On August 19, 2009 (i.e., ADAMS Accession Number ML092321080), CP&L presented plans to use the ATRIUM-10XM (A10XM) fuel design at BSEP to the NRC. As presented on August 19, 2009, use of the A10XM fuel design includes the addition of AREVA Topical Report BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008, to the list of analytical methods specified in Technical Specification 5.6.5.b that have been reviewed and approved by the NRC for determining core operating limits. This methodology uses the RODEX4 fuel performance code.

By letter dated April 15, 2010, AREVA provided to the NRC the results of evaluations performed for the A10XM fuel design to demonstrate compliance with NRC approved fuel licensing criteria defined in ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995. These results are presented in ANP-2899P, Revision 0, *Fuel Design Evaluation for ATRIUM™ 10XM BWR Reload Fuel*, provided as an enclosure to the April 15, 2010 AREVA letter. With ANF-89-98(P)(A) Revision 1 and Supplement 1, the NRC approved a set of generic acceptance criteria to be satisfied by AREVA for new BWR fuel designs. In accordance with the process described in ANF-89-98(P)(A) Revision 1 and Supplement 1, new fuel designs or fuel design changes satisfying the ANF-89-98(P)(A) acceptance criteria do not require explicit staff review and approval (i.e., satisfaction of the acceptance criteria is sufficient for approval by reference to the acceptance criteria). Note that, as stated in Section 1.0 of BAW-10247(P), Revision 0, the BAW-10247PA, Revision 0, methodology includes some modifications to the fuel rod criteria in ANF-89-98(P)(A) Revision 1 and Supplement 1, based on the approval and application of the RODEX4 code.

In a separate license amendment request, submitted by letter dated April 29, 2010, CP&L has proposed another revision to TS 5.6.5.b to incorporate an unrelated AREVA topical

topical report (i.e., AREVA Topical Report ANP-10298PA, *ACE/ATRIUM 10XM Critical Power Correlation*, Revision 0, March 2010). The proposed addition of AREVA Topical Reports BAW-10247PA, Revision 0, and ANP-10298PA, Revision 0, are independent of each other.

4.0 Technical Analysis

System Description/Applicable Safety Analysis

Currently, reactor core linear heat generation rate limits are determined using the analytical methodology described in Topical Reports XN-NF-81-58(P)(A), *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model* and EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*. These topical reports have been previously accepted by the NRC and are listed in Technical Specification 5.6.5.b as methodologies that may be used to determine core operating limits for BSEP, Units 1 and 2. The proposed amendments will add AREVA Topical Report BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008, to the list of analytical methods specified in Technical Specification 5.6.5.b that have been reviewed and approved by the NRC for determining core operating limits.

Upon approval of this license amendment application and incorporation of Topical Report BAW-10247PA, Revision 0, into the BSEP Unit 1 and 2 Technical Specifications, CP&L will implement the analytical methods described in the report and in conformance with the limitations described in the topical report and the NRC's safety evaluation. BAW-10247PA will be used to calculate reactor core linear heat generation rate limits for the A10XM fuel design.

Conformance with Methodology and Safety Evaluation Limitations

Section 5.2.3 of BAW-10247PA describes the application of power distribution measurement uncertainties (i.e., radial and axial) by the BAW-10247PA methodology. The radial and axial uncertainties are calculated from separately determined uncertainty components as described in EMF-2158(P)(A), *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*. Three of the uncertainty components used to calculate these uncertainties are determined using traversing incore probe (TIP) measurements. These uncertainty components are:

- The deviation between the CASMO-4/MICROBURN-B2 (C4/MB2) calculated TIP response and the measured TIP response on a radial ($\delta T'_{ij}$), nodal ($\delta T'_{ijk}$) and planar ($\delta T'_{planar}$) basis,

- TIP measurement uncertainty on a radial ($\delta T^{m}{}_{ij}$), nodal ($\delta T^{m}{}_{ijk}$) and planar ($\delta T^{m}{}_{planar}$) basis and
- Synthesis uncertainty on a radial (δS_{ij}) and nodal (δS_{ijk}) basis.

The axial and radial uncertainties from EMF-2158(P)(A) are applicable to a plant-specific application if the plant-specific values of the three uncertainty components identified above produce axial and radial uncertainties bounded by those from EMF-2158(P)(A). This will be the case if the plant-specific values for all three uncertainty components identified above are bounded by the values reported in Sections 9.4 and 9.5 of EMF-2158(P)(A), and the net calculated TIP distribution uncertainty components (δT_{ijk}, δT_{ij} and δT_{planar}), calculated in accordance with the method shown by EMF-2158(P)(A) Equation 9-19, are also bounded by the values reported in Section 9.4 of EMF-2158(P)(A).

CP&L provided a similar confirmation, on a radial basis only, for the three TIP measurement based uncertainty components identified above by letter dated February 14, 2008 (i.e., ADAMS Accession Number ML080520349). CP&L has recalculated these uncertainty components on a radial, nodal and planar basis after adding additional BSEP Unit 1 and Unit 2 TIP measurements obtained through February 2010 to the database used in the CP&L letter dated February 14, 2008.

The recalculated BSEP specific uncertainty component values are tabulated below and are bounded by the proprietary D-Lattice values identified in Sections 9.4 and 9.5 of EMF-2158(P)(A). Both BSEP units are D-Lattice plants. The net calculated TIP distribution uncertainty components (δT_{ijk}, δT_{ij} and δT_{planar}), determined in accordance with the method shown by EMF-2158(P)(A) Equation 9-19, are not provided herein to preclude divulging the proprietary calculation method; however, the method shown by Equation 9-19 may be applied, and the results confirm the BSEP-specific net calculated TIP distribution uncertainty components are also less than the D-Lattice values reported in Section 9.4 of EMF-2158(P)(A). Therefore, the axial and radial uncertainties applied by the BAW-10247PA methodology, which are determined in accordance with the EMF-2158(P)(A) methodology, are applicable to BSEP Units 1 and 2.

Component	BSEP Specific Value
$\delta T'_{ijk}$	4.47%
$\delta T'_{ij}$	2.07%
$\delta T'_{planar}$	2.58%
δS_{ij}	0.22%

Component	BSEP Specific Value
$\delta T^{m}{}_{ijk}$	1.90%
$\delta T^{m}{}_{ij}$	1.25%
$\delta T^{m}{}_{planar}$	1.97%
δS_{ijk}	1.79%

As explained in the CP&L letter dated February 14, 2008, it is not meaningful to trend TIP measurement or synthesis uncertainties versus core conditions. Therefore, only deviations between the C4/MB2 calculated TIP response and the measured TIP response on a radial ($\delta T'_{ij}$), nodal ($\delta T'_{ijk}$) and planar ($\delta T'_{planar}$) basis are plotted versus core power, core average void fraction and core power to core flow ratio in Figures 1 through 3 of this enclosure to illustrate the BSEP TIP database. This presentation is consistent with that previously requested by the NRC and provided by CP&L in the CP&L letter dated February 14, 2008, for $\delta T'_{ij}$ alone. Figures 1 through 3 of this Enclosure 1 present the TIP data versus core void calculated by MICROBURN-B2, whereas the CP&L letter dated February 14, 2008 presented TIP data versus core void calculated by the POWERPLEX-II core monitoring system then in use. There is no material impact on core void trends using either basis; however the difference is noted, because small differences in the void basis are apparent in comparison to the plots provided in the CP&L letter dated February 14, 2008.

The impact of channel bow on the mechanical criteria evaluated with RODEX4 is accounted for by the BAW-10247PA methodology. The channel bow model and its uncertainty are described in Appendix B of BAW-10247PA. The model is implemented in the MICROBURN-B2 core simulator. Model uncertainty is based on AREVA fuel channel measurements. Channel fluence is an input to the channel bow model, and is calculated by the MICROBURN-B2 core simulator based on BSEP-specific reactor core operating conditions. AREVA fuel loaded in the BSEP Unit 1 and 2 reactor cores is channeled with AREVA fuel channels made of Zircaloy-4 material. Operating experience has shown Zircaloy-4 channel material is not susceptible to abnormal shadow corrosion enhanced channel bow. The channel bow model described in Appendix B of BAW-10247PA is applicable to BSEP, because the model accounts for channel fluence calculated specific to the BSEP reactor core operating conditions. The channel bow model uncertainty is applicable to BSEP, because Zircaloy-4 AREVA fuel channels used by BSEP are not susceptible to abnormal channel bow.

The NRC identified five limitations and conditions on use of the BAW-10247PA methodology. These limitations and conditions, and the demonstration that BSEP complies with them, follow.

Limitation and Condition 1:
"Due to limitations within the FGR model, the analytical fuel pellet grain size shall not exceed 20 microns 3-D when the as-manufactured fuel pellet grain size could exceed 20 microns 3-D."

BSEP core operating limits analyses performed with the BAW-10247PA, Revision 0 methodology will not use an analytical fuel pellet grain size in excess of 20 microns 3-D.

Limitation and Condition 2:
"RODEX4 shall not be used to model fuel above incipient fuel melting temperatures."

BSEP core operating limits analyses performed with RODEX4 will not model fuel above incipient fuel melting temperatures.

Limitation and Condition 3:
"The hydrogen pickup model within RODEX4 is not approved for use."

BSEP core operating limits analyses performed with RODEX4 will not use the hydrogen pickup model within RODEX4.

Limitation and Condition 4:
"Due to the empirical nature of the RODEX4 calibration and validation process, the specific values of the equation constants and tuning parameters derived in TR BAW-10247(P), Revision 0, (as updated by RAI responses) become inherently part of the approved models. Thus, these values may not be updated without necessitating further NRC review."

BSEP core operating limits analyses performed with the BAW-10247PA, Revision 0 methodology will use the specific values of the equation constants and tuning parameters derived in Topical Report BAW-10247(P), Revision 0 (as updated by RAI responses).

Limitation and Condition 5:
"RODEX4 has no crud deposition model. Due to the potential impact of crud formation on heat transfer, fuel temperature, and related calculations, RODEX4 calculations must account for a design basis crud thickness. The level of deposited crud on the fuel rod surface should be based upon an upper bound of expected crud and may be based on plant-specific history. Specific analyses would be required if an abnormal crud or corrosion layer (beyond the design basis) is observed at any given plant. For the purpose of this evaluation, an abnormal crud/corrosion layer is defined by a formation that increases the calculated fuel average temperature by more than 25°C beyond the design basis calculation."

BSEP core operating limits analyses performed with the BAW-10247PA, Revision 0 methodology will account for a design basis crud thickness as described on Page 4 of 4 of the proprietary file "Initial RODEX4 Draft SER Comments.doc," attached to eMail dated November 2, 2007, from J. S. Holm (AREVA) to Holly Cruz (NRC), and incorporated in the approved BAW-10247PA, Revision 0, Topical Report. As described in Section 3.3 of the NRC SER approving BAW-10247(P), the BAW-10247PA methodology bounds corrosion data with greater than 95/95 confidence.

Abnormal BSEP Unit 1 or Unit 2 fuel crud and corrosion have not been observed and are not expected. Inspections of irradiated GE14 fuel operated in BSEP Unit 1 and Unit 2 have consistently found the fuel to be clean with no evidence of tenacious crud or enhanced cladding corrosion. Low crud levels and generally cleaner condition compared to other plants with AREVA fuel were similarly noted during recently completed inspections of irradiated ATRIUM-10 fuel operated in BSEP Unit 1 Cycle 17. BSEP operating procedures require that sampling and analysis of water chemistry parameters be consistent with the guidance in the latest approved revision of the EPRI Water Chemistry Guidelines (i.e., currently *BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines—2008 Revision.* EPRI, Palo Alto, CA: 2008. 1016579).

Topical Report BAW-10247PA identifies three additional conditions delineated in the conclusion of the NRC SER for BAW-10247(P). These additional conditions, and the demonstration that BSEP complies with them, follow.

<u>BAW-10247(P) Condition 1</u>:
RODEX4 is approved for modeling BWR fuel rods with a peak rod average burnup limit of 62 GWd/MTU.

RODEX4 will not be used to calculate BSEP core operating limits for fuel rods with a peak rod average burnup greater than 62 GWd/MTU.

<u>BAW-10247(P) Condition 2</u>:
RODEX4 is approved for modeling BWR fuel rods consisting of a solid UO2 fuel pellet with a maximum gadolinia content of 10.0 weight percent.

BSEP core operating limits analyses performed with RODEX4 will not model BWR fuel rods consisting of other than a solid UO2 fuel pellet with a maximum gadolinia content of 10.0 weight percent.

<u>BAW-10247(P) Condition 3</u>:
RODEX4 is approved for modeling BWR fuel rods with CWSR Zr-2 fuel clad material.

BSEP core operating limits analyses performed with RODEX4 will not model BWR fuel rods with other than CWSR Zr-2 fuel clad material.

ATRIUM-10XM Fuel Design

The NRC approved the use of AREVA fuel and core design methodologies to determine BSEP core operating limits with the issuance of License Amendments 246 and 274 for BSEP, Units 1 and 2, respectively. AREVA licensing topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 is one of these NRC-approved methodologies. ANF-89-98(P)(A) Revision 1 and Supplement 1, as clarified by a Siemens Power Corporation letter dated October 12, 1999 (i.e., Reference 1) and an NRC letter dated May 31, 2000 (i.e., Reference 2) requires that a summary of the evaluation of the A10XM design against the NRC-approved generic design criteria be provided to the NRC for information. AREVA provided this evaluation to the NRC for information by letter dated April 15, 2010, which transmitted AREVA document ANP-2899P, Revision 0, *Fuel Design Evaluation for ATRIUM™ 10XM BWR Reload Fuel.* In accordance with the process described in ANF-89-98(P)(A) Revision 1 and Supplement 1, new fuel designs or fuel design changes satisfying the ANF-89-98(P)(A) design criteria do not require explicit NRC review and approval (i.e., satisfaction of the design criteria is sufficient for approval by reference to the criteria).

ANP-2899P identifies fuel design criteria, specified in ANF-89-98(P)(A) Revision 1 and Supplement 1, which are evaluated on a cycle-specific basis. Fuel rod criteria are also included in BAW-10247PA with the application of RODEX4. Reports summarizing the results of analyses performed to demonstrate BSEP compliance with the cycle-specific criteria are provided by AREVA to CP&L as part of the normal reload licensing document package. This type of information is not available until later in the reload licensing process. Consistent with the process described in ANF-89-98(P)(A) Revision 1 and Supplement 1 (as clarified by References 1 and 2), CP&L will provide the reports produced for the BSEP Unit 2 Cycle 20 reload to the NRC for information. The reports will be provided in supplemental letters as they are completed during the reload licensing process, on the schedule presented below.

Report	Schedule for Transmitting to NRC
Fuel Cycle Design Report	August 2010
Thermal-Hydraulic Design Report	August 2010
LOCA Analysis Reports	September 2010
Mechanical Design Reports	November 2010
Reload Safety Analysis Report	November 2010

ANP-2899P also identifies fuel design criteria, specified in ANF-89-98(P)(A) Revision 1 and Supplement 1, that are evaluated on a plant-specific basis. These criteria address thermal hydraulic compatibility, fuel lift-off, structural deformation and LOCA performance.

The key differences in system configuration between BSEP Unit 1 and Unit 2 are in the core inlet region and the Turbine Bypass System. The orifice diameter in Unit 2 is smaller than Unit 1, 2.09 inches compared to 2.43 inches, and the Turbine Bypass System for Unit 2 has 10 valves whereas Unit 1 has 4 valves. Differences in neutronic design and operation are minimal since both units operate on 24 month fuel cycles.

Based on the minimal differences between Units 1 and 2, CP&L will include, for information, the Thermal-Hydraulic Design and Reload Safety Analysis Reports with our transmittal of the Core Operating Limits Report prior to startup from the first Unit 1 refueling outage that loads AREVA A10XM fuel into the reactor core. These reports will summarize compliance with plant and cycle specific ANF-89-98(P)(A) Revision 1 and Supplement 1 fuel design criteria for the first Unit 1 cycle that uses A10XM fuel. Additional information supporting evaluation of plant-specific thermal-hydraulic compatibility, fuel lift-off, structural deformation and LOCA performance criteria for Unit 1 is provided below.

Section 4.1.1 of ANP-2899P presents the results of an example thermal-hydraulic compatibility analysis to demonstrate that the A10XM fuel is compatible with the ATRIUM-10 fuel design for an example BWR/4 core. The example BWR/4 core used for this analysis is BSEP Unit 1; however, the Thermal-Hydraulic Design Report prepared for A10XM fuel introduction into BSEP Unit 1 will also be provided to the NRC for information as described above.

The plant-specific fuel lift off criteria are evaluated for A10XM as described in Section 3.3.8 of ANP-2899P. These criteria are satisfied for the ATRIUM-10 fuel design currently operating in both BSEP Unit 1 and Unit 2. A10XM fuel lift margin is greater than ATRIUM-10 based on fuel mass and pressure drop differences presented in Tables 2.1 and 4.2 of ANP-2899P; therefore, plant-specific fuel lift criteria remain satisfied for both BSEP Unit 1 and Unit 2 with A10XM fuel.

The plant-specific structural deformation fuel design criteria are evaluated for A10XM as described in Section 3.4.4 of ANP-2899P. The structural deformation fuel design criteria were confirmed for the ATRIUM-10 fuel design currently operating in both BSEP Unit 1 and Unit 2 based on BSEP core support plate motions calculated using ATRIUM-10 fuel channel properties and an assembly weight slightly greater than the ATRIUM-10 fuel design. These BSEP core support plate motions are not impacted by use of A10XM fuel, because the A10XM fuel design uses channels of the same design and material (i.e., Zircaloy-4) as the ATRIUM-10 fuel design, and the A10XM assembly weight is consistent with that used to calculate the BSEP core support plate motions. AREVA evaluates the consequences of BSEP core support plate motion on A10XM fuel. The results of this AREVA evaluation will be summarized in the Brunswick Unit 2 Cycle 20 Mechanical Design Report, which will be provided to the NRC for information as described above. These results will be applicable to both BSEP Unit 1 and Unit 2, because the BSEP Unit 1 and Unit 2 core support plate motions are the same.

The plant-specific LOCA performance criteria are evaluated for A10XM as described in ANP-2899P. Break spectrum and heat-up LOCA analyses are performed as part of the reload analyses for the first reload using a new fuel design, and only heat-up analyses are performed for any new lattice designs introduced thereafter. Heat-up analyses performed for new lattice designs typically confirm the applicability of the initial LOCA analyses. The bounding BSEP unit is determined and analyzed as part of the initial reload LOCA analyses, so that the analyses will be applicable to both BSEP Unit 1 and Unit 2. The results of the initial reload evaluation will be summarized in the Brunswick LOCA Reports, which will be provided to the NRC for information as described above.

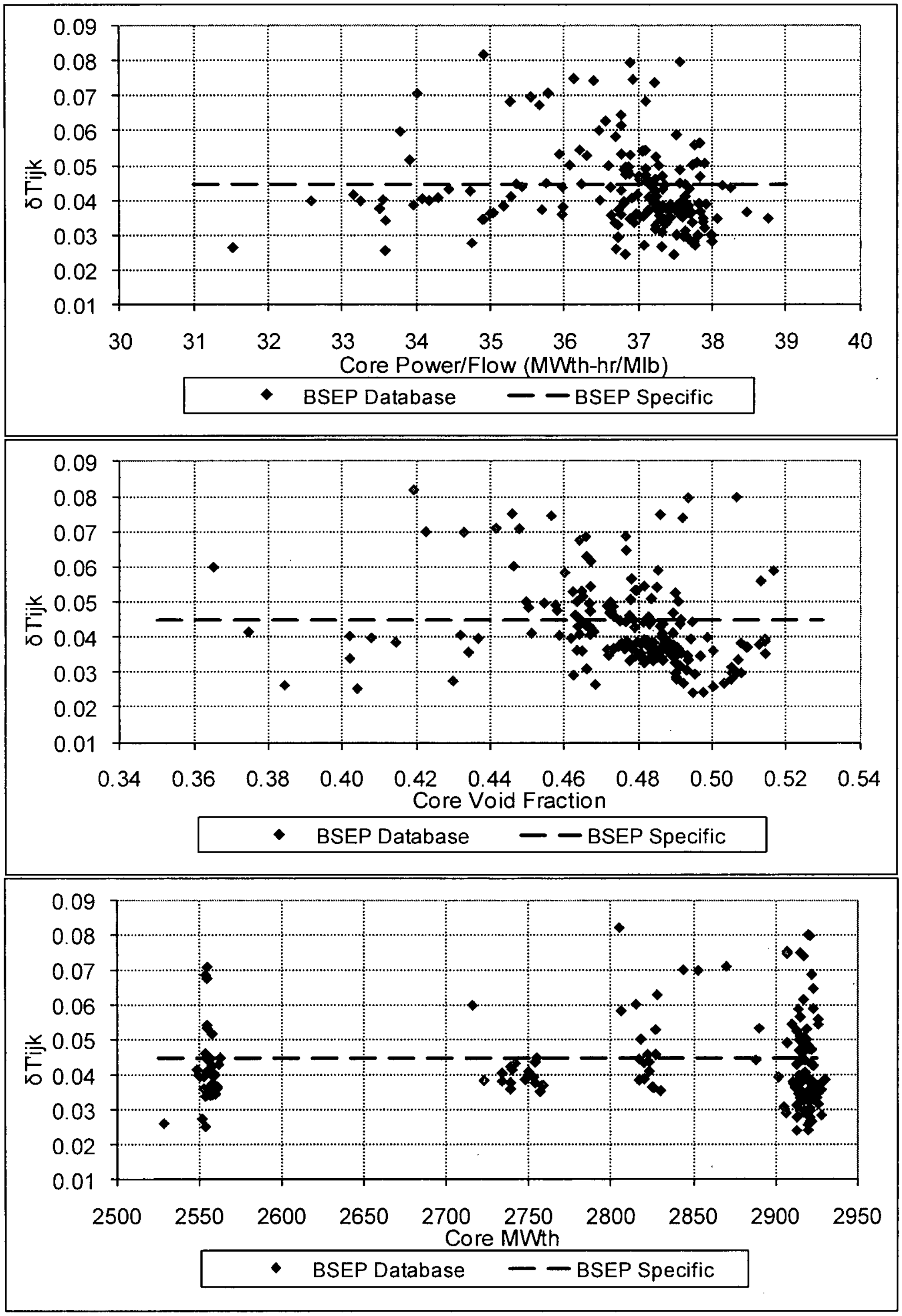

Figure 1: BSEP δT'ijk Gamma TIP Database

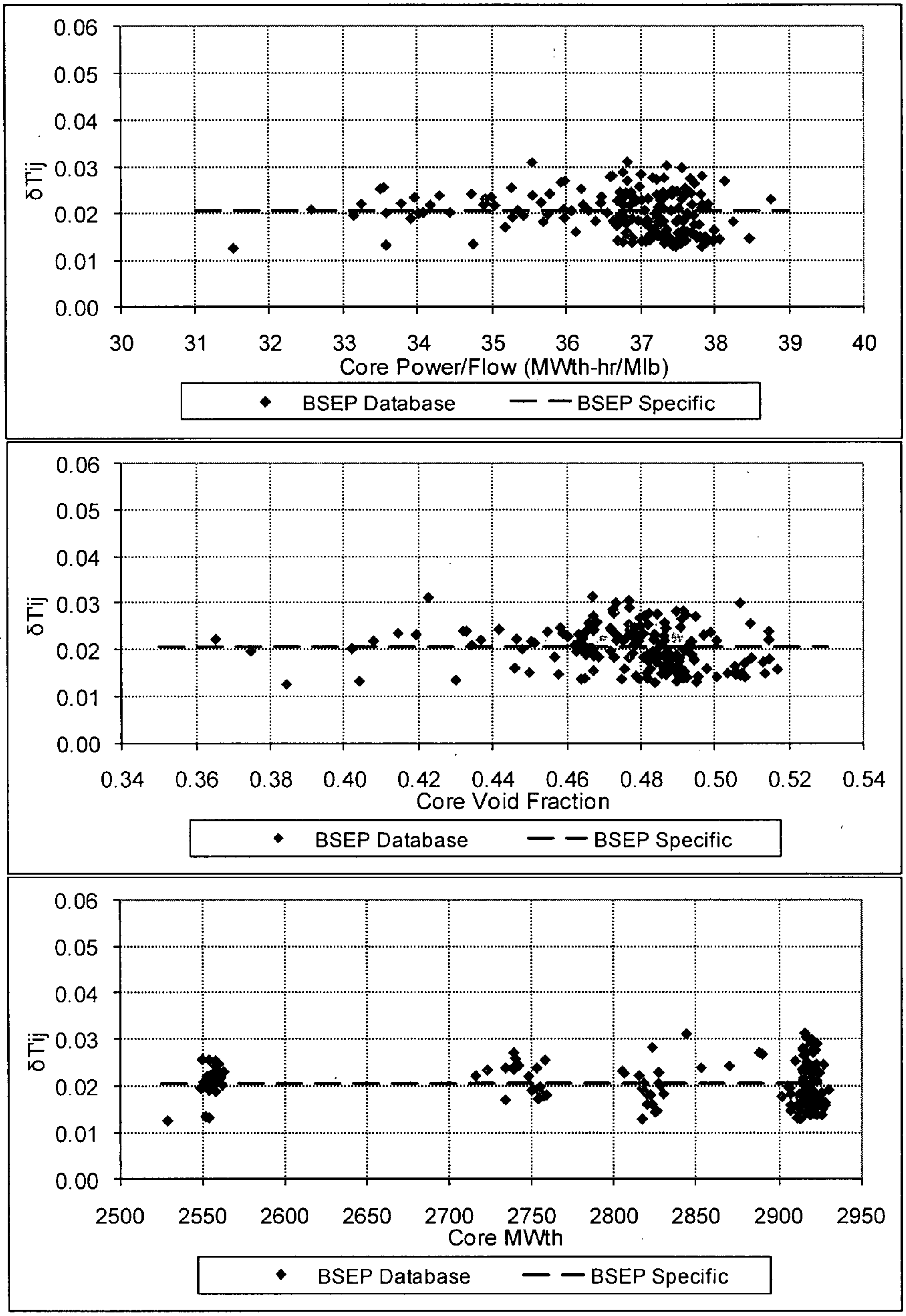

Figure 2: BSEP δT'ij Gamma TIP Database

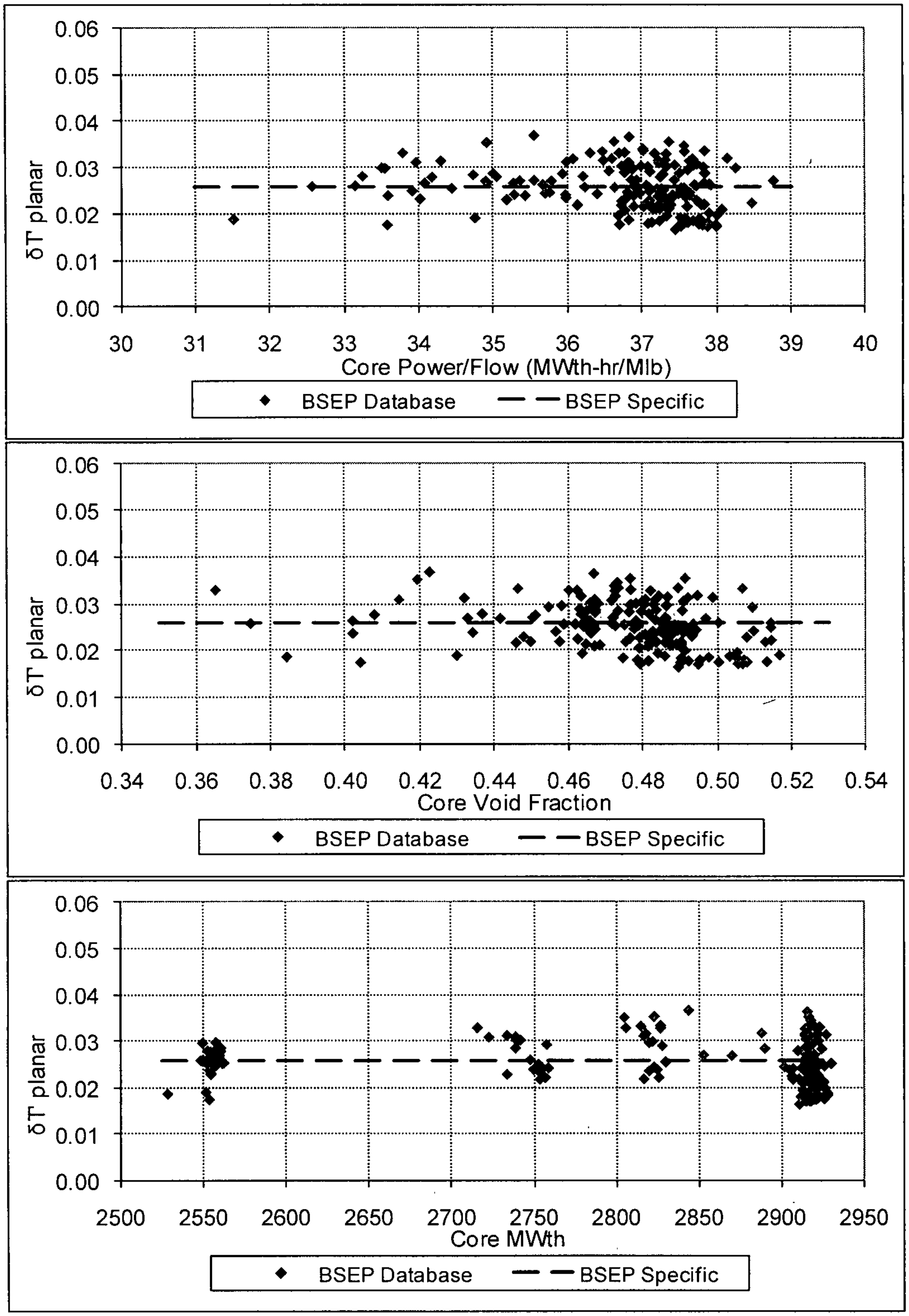

Figure 3: BSEP δT'planar Gamma TIP Database

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

The proposed change will add, to Technical Specification 5.6.5.b, an additional topical report describing an NRC reviewed and approved analytical method for determining core operating limits. The new analytical method, which is in AREVA Topical Report BAW-10247PA, Revision 0, describes a statistical thermal-mechanical evaluation methodology for boiling water reactor fuel rods based on the RODEX4 best-estimate fuel performance code. CP&L has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods identified in Technical Specification 5.6.5.b that can be used to establish core operating limits. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Since no individual precursors of an accident are affected, the proposed amendments do not increase the probability of a previously analyzed event.

   The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods used to establish core operating limits. The addition of the topical report to Technical Specification 5.6.5.b will allow a new thermal-mechanical methodology, based on the RODEX4 fuel performance code, to be used to determine reactor core linear heat generation rate limits monitored as specified by Technical Specification 3.2.3. The addition of the analytical methodology described in Topical Report BAW-10247PA to Technical Specification 5.6.5.b does not alter the assumptions of accident analyses or the Technical Specification Bases. Based on the above, the proposed amendments do not increase the consequences of a previously analyzed accident.

   Therefore, the proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed amendments do not involve any plant configuration modifications or changes to allowable modes of operation. The proposed Topical Report addition to Technical Specification 5.6.5.b provides an analytical methodology for determining reactor core linear heat generation rate limits that ensures no new accident precursors are created. Therefore, the proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods identified in Technical Specification 5.6.5.b that can be used to establish core operating limits. This addition to Technical Specification 5.6.5.b will allow a new NRC-accepted analytical methodology to be used to determine reactor core linear heat generation rate limits.

   Limits on the linear heat generation rate are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences. Exceeding the linear heat generation rate limit could potentially result in fuel damage and subsequent release of radioactive materials. The mechanisms that could cause fuel damage during normal operations and operational transients and that are considered in fuel evaluations are rupture of the fuel rod cladding caused by strain and overheating of the fuel. The proposed change will ensure the current level of fuel protection is maintained (i.e., that the fuel design safety criteria of less than one percent plastic strain of the fuel cladding is met and incipient centerline melting of the fuel does not occur) and thus assure that rupture of the fuel rod cladding caused by strain and overheating of the fuel does not occur.

   Therefore, the proposed amendments do not result in a significant reduction in the margin of safety.

Based on the above, CP&L concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

CP&L has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR).

10 CFR 50.36(c)(5) states that the Technical Specifications will include administrative controls that address the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The Core Operating Limits Report (COLR) is required as a part of the reporting requirements specified in the Brunswick Technical Specifications Administrative Controls section. The Technical Specifications requires the core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and to be documented in the COLR. In addition, it requires the analytical methods used to determine the core operating limits to be those that have been previously reviewed and approved by the NRC, and specifically to be those described in Technical Specification 5.6.5.b. The proposed amendments ensure that these requirements are met.

Generic Letter (GL) 88-16, *Removal of Cycle-Specific Parameters from Technical Specifications*, provided guidance on relocating numerical values in Technical Specifications and the referencing of associated methodology topical reports in the Administrative Controls Technical Specifications and the COLR. However, in a recent letter to the Technical Specification Task Force (TSTF) (i.e., Reference 5), the NRC has informed the TSTF that relaxation of Technical Specification Topical Report documentation from the guidance provided in GL 88-16 and Technical Specification Task Force (TSTF) Traveler TSTF-363, *Revise Topical Report References in ITS 5.6.5, COLR*, is no longer appropriate. Accordingly, the proposed addition of the AREVA Topical Report BAW-10247PA methodology reference to Technical Specification 5.6.5.b includes the information necessary to identify the specific revision of the methodology (i.e., the revision number and issuance date).

10 CFR 50, Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems 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.

To ensure compliance with GDC 10, linear heat generation rate analyses are performed using NRC-approved methodologies. Limits on the linear heat generation rate are specified to ensure that fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences.

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

6.0 Environmental Considerations

A review has determined that the proposed amendment is administrative in nature and does not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and does not change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

1. Letter from James F. Mallay (Siemens Power Corporation) to U.S. Nuclear Regulatory Commission, *Revision to Attachment 1 of letter NRC:99:030, Request for Concurrence on SER Clarifications*, October 12, 1999.

2. Letter from Stuart Richards (Siemens Power Corporation) to U.S. Nuclear Regulatory Commission, *Siemens Power Corporation Re: Request for Concurrence on Safety Evaluation Report Clarifications (TAC No. MA6160)*, May 31, 2000.

3. Letter from Farideh E. Saba (USNRC) to Benjamin Waldrep (CP&L), *Issuance of Amendments to Support Transition to AREVA Fuel and Methodologies (TAC*

Nos. MD4063 and MD4064), March 27, 2008, ADAMS Accession Number ML080870478.

4. Letter from James Scarola (CP&L) to U.S. Nuclear Regulatory Commission, *Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel*, January 22, 2007, ADAMS Accession Number ML070300570.

5. Letter from Stacey L. Rosenberg (NRC) to the Technical Specification Task Force, "Technical Specification Task Force (TSTF) Traveler 363, Revision 0, 'Revise Topical Report References in ITS 5.6.5, COLR'," dated November 2, 2009.

Marked-up Technical Specification Pages - Unit 1

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

(continued)

Typed Technical Specification Pages - Unit 1

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

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.

5.6.6 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.

Typed Technical Specification Page - Unit 2

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

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.

5.6.6 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.