Southern Nuclear

B. H. Whitley
Director
Regulatory Affairs

Southern Nuclear
Operating Company, Inc.
42 Inverness Center Parkway
Birmingham, AL 35242
Tel 205.992.7079
Fax 205.992.5296

February 2, 2018

Docket Nos.: 52-025
52-026

ND-18-0057
10 CFR 50.90

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

**Southern Nuclear Operating Company**
**Vogtle Electric Generating Plant Units 3 and 4**
**Request for License Amendment Regarding Analytical Methods for**
**Core Operating Limits Report and Consistency Changes (LAR-18-006)**

Ladies and Gentlemen:

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the combined licenses (COLs) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 (License Numbers NPF-91 and NPF-92, respectively). The proposed amendment would revise the licensing basis information regarding the following:

- Administrative changes to COL Appendix A Technical Specification (TS) 5.6.3 for the core operating limits report (COLR) required documentation to include analytical methods which are described elsewhere in the TS and in the Updated Final Safety Analysis Report (UFSAR), and

- An editorial change to COL Appendix A TS 5.7.2 for high radiation areas to correct a typographical error.

The requested amendment also proposes changes to depart from approved AP1000 Design Control Document (DCD) Tier 2 information (text and tables) as incorporated into the UFSAR as plant-specific DCD information, and proposes to depart from Plant-Specific Technical Specifications (PS-TS) as incorporated in Appendix A of the COL.

Enclosure 1 provides the description, technical evaluation, regulatory evaluation (including the significant hazards consideration determination), and environmental considerations for the proposed changes.

Enclosure 2 provides markups depicting the requested changes to the VEGP Units 3 and 4 licensing basis documents.

This letter, including enclosures, has been reviewed and confirmed to not contain security-related information. This letter contains no regulatory commitments.

SNC requests NRC staff review and approval of the license amendment no later than February 1, 2019. Approval by this date will allow sufficient time to implement licensing basis changes prior to VEGP 3 and 4 commencing operation. SNC expects to implement the proposed amendment [through incorporation into the licensing basis documents (e.g., the COLR and UFSAR)] within 30 days of approval of the requested changes.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia by transmitting a copy of this letter and its enclosures to the designated State Official.

Should you have any questions, please contact Ms. Amy Chamberlain at (205) 992-6361.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd of February 2018.

Respectfully submitted,

B. H. Whitley

Brian H. Whitley
Director, Regulatory Affairs
Southern Nuclear Operating Company

Enclosures 1) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 - Request for License Amendment Regarding Analytical Methods for Core Operating Limits Report and Consistency Changes (LAR-18-006)

2) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 - Proposed Changes to the Licensing Basis Documents (LAR-18-006)

cc:

Southern Nuclear Operating Company / Georgia Power Company
Mr. S. E. Kuczynski (w/o enclosures)
Mr. M. D. Rauckhorst
Mr. D. G. Bost (w/o enclosures)
Mr. M. D. Meier (w/o enclosures)
Mr. D. H. Jones (w/o enclosures)
Mr. D. L. McKinney (w/o enclosures)
Mr. T. W. Yelverton (w/o enclosures)
Mr. B. H. Whitley
Mr. J. J. Hutto
Mr. C. R. Pierce
Ms. A. G. Aughtman
Mr. D. L. Fulton
Mr. M. J. Yox
Mr. J. Tupik
Mr. W. A. Sparkman
Ms. A. C. Chamberlain
Ms. A. L. Pugh
Mr. J. D. Williams
Mr. F. J. Redwanz
Document Services RTYPE: VND.LI.L00
File AR.01.02.06

Nuclear Regulatory Commission
Mr. W. Jones (w/o enclosures)
Ms. J. Dixon-Herrity
Mr. C. Patel
Ms. J. M. Heisserer
Mr. B. Kemker
Mr. G. Khouri
Ms. S. Temple
Mr. F. Brown
Mr. T.E. Chandler
Ms. P. Braxton
Mr. T. Brimfield
Mr. C. J. Even
Mr. A. Lerch

State of Georgia
Mr. R. Dunn

<u>Oglethorpe Power Corporation</u>
Mr. M. W. Price
Mr. K. T. Haynes
Ms. A. Whaley

<u>Municipal Electric Authority of Georgia</u>
Mr. J. E. Fuller
Mr. S. M. Jackson

<u>Dalton Utilities</u>
Mr. T. Bundros

<u>Westinghouse Electric Company, LLC</u>
Mr. L. Oriani (w/o enclosures)
Mr. G. Koucheravy (w/o enclosures)
Mr. M. Corletti
Mr. M. L. Clyde
Ms. L. Iller
Mr. D. Hawkins
Mr. J. Coward

<u>Other</u>
Mr. S. W. Kline, Bechtel Power Corporation
Ms. L. A. Matis, Tetra Tech NUS, Inc.
Dr. W. R. Jacobs, Jr., Ph.D., GDS Associates, Inc.
Mr. S. Roetger, Georgia Public Service Commission
Ms. S. W. Kernizan, Georgia Public Service Commission
Mr. K. C. Greene, Troutman Sanders
Mr. S. Blanton, Balch Bingham
Mr. R. Grumbir, APOG
NDDocumentinBox@duke-energy.com, Duke Energy
Mr. S. Franzone, Florida Power & Light

**Southern Nuclear Operating Company**

**ND-18-0057**

**Enclosure 1**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Request for License Amendment Regarding**
**Analytical Methods for Core Operating Limits Report and Consistency Changes**

**(LAR-18-006)**

(This Enclosure consists of 19 pages, including this cover page)

**Table of Contents**

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to Combined License (COL) Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

## 1. SUMMARY DESCRIPTION

The proposed amendment would revise the licensing basis information regarding the following:

- Administrative changes to COL Appendix A Technical Specification (TS) 5.6.3 for the core operating limits report (COLR) required documentation to include analytical methods which are described elsewhere in the TS and in the Updated Final Safety Analysis Report (UFSAR), and
- An editorial change to COL Appendix A TS 5.7.2 for high radiation areas to correct a typographical error.

The requested amendment proposes changes to depart from approved AP1000 Design Control Document (DCD) Tier 2 information (text and tables) as incorporated into the UFSAR as plant-specific DCD information (as detailed in Section 2), and proposes to depart from Plant-Specific Technical Specifications (PS-TS) as incorporated in Appendix A of the COL.

## 2. DETAILED DESCRIPTION and TECHNICAL EVALUATION

### 2.1 Changes to COL Appendix A TS 5.6.3, CORE OPERATING LIMITS REPORT (COLR)

TS 5.6 identifies reports which must be submitted in accordance with 10 CFR 50.4. TS 5.6.3.a identifies core operating limits which must be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and documented in the COLR. TS 5.6.3.b identifies the analytical methods previously reviewed and approved by the NRC used to determine the core operating limits. TS 5.6.3.a and TS 5.6.3.b are proposed to be revised to be consistent with the requirements which are described elsewhere in the TS and in the UFSAR, including the NRC-approved analytical methods used in core design for VEGP Units 3 and 4 as described below.

<u>Removal of Reactor Trip System Instrumentation from COLR</u>

TS 5.6.3.a identifies TS 3.3.1, Reactor Trip System (RTS) Instrumentation, as one of the core operating limits which must be established prior to each reload cycle and which must be documented in the COLR. The Overpower Delta-T (OPΔT) and Overtemperature Delta-T (OTΔT) reactor trip setpoints and time constants were previously planned to be included in the COLR to support TS 3.3.1. However, the OPΔT and OTΔT reactor trip setpoints and time constants have since been included in the TS 5.5.14 Setpoint Program (SP). This impact to TS 5.6.3 was not recognized at the time of the change.

As required by TS 5.5.14.e, the SP shall establish a document containing the current value of the specified Nominal Trip Setpoint (NTS), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) for each TS required automatic protection instrumentation function and references to the calculation documentation. Changes to this document shall be governed by the regulatory requirement of 10 CFR 50.59. In addition, changes to the specified NTS, AFT, and ALT values shall be governed by the approved setpoint methodology specified in

TS 5.5.14.b. This document, including any revisions or supplements, shall be provided upon issuance to the NRC.

Additionally, as required by TS 5.5.14.b, each TS required automatic protection instrumentation setpoint and tolerance shall be calculated in conformance with WCAP-16361-P-A and WCAP-16361-NP-A (Reference 6.1). The time constants and equations used for the OPΔT and OTΔT reactor trips are discussed in both WCAP-16361-P-A/WCAP-16361-NP-A and APP-GW-GLR-137 (Reference 6.2). APP-GW-GLR-137 contains discussion regarding the basis for the OPΔT and OTΔT reactor trip algorithms, and both WCAP-16361-P-A and TS Bases 3.3.1 describe APP-GW-GLR-137 as the appropriate analytical method for the revised OPΔT and OTΔT reactor trips.

Therefore, there is no need for the COLR to contain the OPΔT and OTΔT reactor trip setpoints and time constants that are already included in the SP document. As a result, it is proposed to remove TS 3.3.1 from the list of the core operating limits which must be established prior to each reload cycle and which must be documented in the COLR. Additionally, TS 3.3.1 is proposed to be removed from the supporting analytical methods discussed in item 6 of TS 5.6.3.b, since TS 3.3.1 limits are no longer contained in the COLR. The reference to APP-GW-GLR-137 is maintained as the analytical method for OPΔT and OTΔT reactor trips, therefore no additional text is added to TS 5.5.14.

These proposed changes do not affect any Limiting Conditions for Operation (LCO), Action, or Surveillance Requirement (SR) as discussed in the TS. The OPΔT and OTΔT reactor trip setpoints and time constants continue to be established for both the initial core design, and for the reload core designs, in the SP as required by TS 5.5.14.b. The UFSAR Chapter 15 safety analyses are not impacted by this change, because the change does not affect any of the inputs, methodology, assumptions, or acceptance criteria that are modeled in the safety analyses since the OPΔT and OTΔT reactor trip setpoints and time constants remain controlled by the SP. No changes to the UFSAR result from these proposed changes. Therefore, these proposed changes are acceptable for maintaining reactor trip setpoints.

WCAP-9272-P-A and WCAP-9273-NP-A Analytical Method TS Applicability

TS 3.1.1, Shutdown Margin (SDM), and TS 3.4.1, Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, are listed as core operating limits which must be established and documented in the COLR in TS 5.6.3.a. However, neither core operating limit is tied to an analytical method discussed in TS 5.6.3.b. Therefore, the TS do not identify the appropriate analytical method used in the COLR to determine SDM and RCS pressure, temperature, and flow DNB limits.

SDM and RCS pressure, temperature, and flow DNB limits are determined in the COLR in accordance with the NRC-approved analytical methods described in WCAP-9272-P-A and WCAP-9273-NP-A (Reference 6.3). The analytical method used to determine SDM is described in Subsection 3.3.2 of both topical reports, and the analytical method used to determine the RCS pressure, temperature, and flow DNB limits are described in Subsection 2.3 of both topical reports. These analytical methods are described in the core design basis of UFSAR Subsection 4.3.1, and have been previously approved by the NRC for determination of SDM and RCS pressure, temperature, and flow DNB as described in UFSAR Subsection 4.3. Therefore, it is proposed to add TS 3.1.1 and TS 3.4.1 to the list of

TS supported by WCAP-9272-P-A and WCAP-9273-NP-A in item 1 of TS 5.6.3.b. This change implements NRC-approved analytical methods WCAP-9272-P-A and WCAP-9273-NP-A for the appropriate COLR items identified in TS 5.6.3.

These proposed changes do not affect any LCO, Action, or SR as discussed in the TS. These reactivity control and RCS core operating limits continue to be established in accordance with the analytical methods as described in the COLR for both the initial core design, and for the reload core designs, as described in UFSAR Subsection 4.3. In addition, no changes are needed to TS 3.1.1 and TS 3.4.1 Bases, because the discussions are already consistent with the use of the analytical methods described in WCAP-9272-P-A and WCAP-9273-NP-A. The UFSAR Chapter 15 safety analyses are not impacted by this change, because the change does not affect any of the inputs, methodology, assumptions, or acceptance criteria that are modeled in the safety analyses since these analytical methods were previously approved by the NRC, and are currently used in the licensing basis as described in UFSAR Subsection 4.3, for establishing these core operating limits. The only change required to the UFSAR is to add these topical reports to UFSAR Table 1.6-1 for DCD Section Number 16.1 for consistency with the TS change described above. Therefore, these proposed changes are acceptable for maintaining reactivity control and RCS core operating limits.

<u>WCAP-16009-P-A and APP-GW-GLE-026 Analytical Method TS Applicability</u>

TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology), requires the heat flux hot channel factor to be within the limits specified in the COLR. TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor, requires the nuclear enthalpy rise hot channel factor to be within the limits specified in the COLR. Complying with the requirements of these TS maintains these hot channel factor power distribution limits within the limits of the COLR to prevent peak fuel cladding temperature exceeding 2200°F during a large-break loss-of-coolant accident (LBLOCA).

WCAP-16009-P-A and WCAP-16009-NP-A (Reference 6.4) describe the Automated Statistical Treatment of Uncertainty Method (ASTRUM) analytical method used to perform the best-estimate LBLOCA analysis as described in the current UFSAR Chapter 15 safety analysis. These changes were incorporated generically for the AP1000 design in the NRC-approved AP1000 core reference report WCAP-17524-P-A, Revision 1 (Reference 6.5), and incorporated into the VEGP Units 3 and 4 licensing basis as Amendments 52 and 52, respectively (NRC Accession Number ML16201A435). Therefore, these analytical methods have been approved by the NRC and have replaced the previous Westinghouse proprietary WCAP-12945-P-A and non-proprietary WCAP-14747 (Reference 6.6) analytical method for best-estimate LBLOCA safety analysis applications.

TS 5.6.3.b Item 4 still lists WCAP-12945-P-A and WCAP-14747 as the analytical method for TS 3.2.1. However, WCAP-16009-P-A and WCAP-16009-NP-A analytical methods have been approved by the NRC and incorporated into the AP1000 design, and the application of these analytical methods is discussed in topical report APP-GW-GLE-026 and APP-GW-GLE-026-NP (Reference 6.7). Therefore, TS 5.6.3.b Item 4 is proposed to be revised to replace WCAP-12945-P-A and WCAP-14747 with WCAP-16009-P-A and WCAP-16009-NP-A as new Item 4a, and with topical reports APP-GW-GLE-026 and APP-GW-GLE-026-NP as new Item 4b, as the analytical methods for TS 3.2.1 to incorporate the NRC-approved analytical methods accepted for use in LBLOCA analyses into the TS.

In addition, TS 3.2.2 is proposed to be added to the list of supported TS for both new Item 4a and Item 4b to incorporate the NRC-approved analytical methods accepted for use in LBLOCA analyses into the TS. The COLR determines this core operating limit in accordance with the WCAP-16009-P-A, WCAP-16009-NP-A, APP-GW-GLE-026, and APP-GW-GLE-026-NP ASTRUM analytical method as described in UFSAR Section 4.3.

In the Final Safety Evaluation (SE) for WCAP-16009-P-A (NRC Accession Number ML043100073), the following conditions and limitations were required by the NRC Staff for using the ASTRUM methodology:

- The ASTRUM methodology only applies to LBLOCA analyses. This condition is met as the ASTRUM methodology is used only in the plant-specific LBLOCA analyses.
- The ASTRUM process of determining the maximum local oxidation and whole core hydrogen generation results is approved, but requires that this information be reported on a plant-specific application which uses ASTRUM. This condition is met as the NRC Staff has approved the application of ASTRUM generically for the AP1000 design in WCAP-17524-P-A, Revision 1, and approved the results of the plant-specific application of ASTRUM as part of the VEGP Units 3 and 4 licensing basis in Amendments 52 and 52, respectively (NRC Accession Number ML16201A435).
- The conditions and limitation previously identified for the Westinghouse computer code used to perform LBLOCA analyses (WCOBRA/TRAC) continue to apply for usage of WCOBRA/TRAC as part of the ASTRUM methodology. This condition is met as the plant-specific LBLOCA analyses continue to use the currently approved application of the WCOBRA/TRAC computer code for the LBLOCA analyses as described in UFSAR Subsection 15.6.5.4A.
- The treatments of the performance criteria of 10 CFR 50.46(b) as addressed in the Code Qualification Document (CQD) methodologies continue to apply unchanged, as previously approved in the SEs for their respective documentation. This condition is met as the plant-specific LBLOCA analyses continue to use the currently approved CQD for the application of the WCOBRA/TRAC computer code for the LBLOCA analyses as described in UFSAR Subsection 15.6.5.4A.1.

These proposed changes do not affect any LCO, Action, or SR as discussed in the TS. These hot channel factor power distribution core operating limits continue to be established in accordance with the COLR for both the initial core design, and for the reload core designs. In addition, no changes are needed to TS 3.2.1 and TS 3.2.2 Bases, because this level of detail is not discussed in the TS Bases regarding the existing uncertainty analyses. The UFSAR Chapter 15 safety analyses are not impacted by this change, because the change does not affect any of the inputs, methodology, assumptions, or acceptance criteria that are modeled in the safety analyses since these analytical methods were previously approved by the NRC, and are currently used in the licensing basis as described in UFSAR Section 4.3, for establishing these core operating limits. The only change required to the UFSAR is to replace the reference to Westinghouse proprietary WCAP-12945-P-A and non-proprietary WCAP-14747 with these topical reports in UFSAR Table 1.6-1 for DCD Section Number 16.1 for consistency with the TS changes described above. Therefore, these proposed changes are acceptable for maintaining the hot channel factor power distribution core operating limits.

WCAP-12472-P-A Analytical Method TS Applicability

The Best Estimate Analyzer for Core Operations – Nuclear (BEACON) system is a core monitoring and support package that uses Westinghouse standard instrumentation in conjunction with an analytical method for online generation of three-dimensional power distributions. The system provides core monitoring, core measurement data reduction, core analysis, and core predictions. WCAP-12472-P-A and WCAP-12473-A, including Addendum 1 and Addendum 2 (Reference 6.8) describe the application of the BEACON system for core monitoring and support. These analytical methods are described in UFSAR Subsection 4.3.2 as used to support core monitoring instrumentation and the experimental verification of power distribution limits, and are also listed in TS 5.6.3.b Item 5 as the appropriate methodologies for TS 3.2.5, On-Line Power Distribution Monitoring System (OPDMS) - Monitored Parameters.

The three-dimensional Advanced Nodal Code (ANC) uses both PARAGON (the Westinghouse NRC-approved lattice code used in core design) and NEXUS (the Westinghouse NRC-approved spatial few-group code used in core design) analytical methods to determine core operating limits, and the related topical reports for PARAGON (Reference 6.9) and NEXUS (Reference 6.10), are described in the core design and safety analysis descriptions in UFSAR Subsections 4.3.3.2, 4.3.3.3, and 15.4.8.2. However, the WCAP-12472-P-A and WCAP-12473-A, Addendum 1, and Addendum 2 analytical methods do not address use of PARAGON and NEXUS as applicable codes used for core design and safety analysis.

WCAP-12472-P-A Addendum 4 and WCAP-12472-NP-A Addendum 4 (Reference 6.11) have two purposes related to the AP1000 design:

1. To affirm the continued use of the NRC-approved Westinghouse design model methodology, including PHOENIX-P/ANC, PARAGON/ANC, and NEXUS/ANC, in the BEACON system.

2. To affirm that uncertainties applied to power distribution monitoring using fixed in-core detectors are valid using higher order polynomial fits of the detector variability and fraction of inoperable detectors than provided in WCAP-12472-P-A and WCAP-12473-A Addendum 1.

The NRC staff has reviewed and approved Revision 0 of WCAP-12472-P-A Addendum 4 and WCAP-12472-NP-A Addendum 4 for referencing in licensing applications using the updated BEACON system, with no restrictions for usage. In the Final SE for WCAP-12472-P-A Addendum 4 (NRC Accession Number ML12158A263), no conditions and limitations were required by the NRC Staff for using the analytical methods described in WCAP-12472-P-A Addendum 4.

Therefore, TS 5.6.3.b Item 5 is proposed to be revised to add WCAP-12472-P-A Addendum 4 and WCAP-12472-NP-A Addendum 4 to the list of analytical methods to incorporate the NRC-approved analytical methods accepted for use in monitoring of core operating limits into the TS. The only changes required to the UFSAR are to add these topical reports to UFSAR Table 1.6-1 for DCD Section Number 4.3 and for DCD Section Number 16.1, and to UFSAR Subsection 4.3.5, Reference 4, for consistency with the

TS changes described above. Therefore, these proposed changes are acceptable for monitoring the power distribution core operating limits.

In addition, a change to the UFSAR is proposed to add reference to WCAP-12472-P-A and WCAP-12473-A Addendum 2 to UFSAR Table 1.6-1 for DCD Section Number 16.1. TS 5.6.3.b, Item 5, lists WCAP-12472-P-A and WCAP-12473-A Addendum 2 as appropriate methodology for OPDMS - Monitored Parameters. However, WCAP-12472-P-A and WCAP-12473-A Addendum 2 are not included in UFSAR Table 1.6-1 for DCD Section Number 16.1. This editorial change aligns the analytical methods identified in UFSAR Table 1.6-1 with the analytical methods used in the COLR as required by the TS.

**Description of any Changes to Current Licensing Basis Documents**

COL Appendix A TS Changes:

The following changes to the COL Appendix A TS are proposed:

1. TS 5.6.3.a is revised to remove 3.3.1, Reactor Trip System (RTS) Instrumentation, from the list of core operating limits required to be established and documented in the COLR.
2. TS 5.6.3.b is revised as follows:
    a. Item 1 is revised to add TS 3.1.1, Shutdown Margin (SDM), and 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, to the list of supported TS for analytical methods WCAP-9272-P-A and WCAP-9273-NP-A.
    b. Item 4 is revised to replace WCAP-12945-P-A with WCAP-16009-P-A and WCAP-16009-NP-A as new Item 4a, and with topical reports APP-GW-GLE-026 and APP-GW-GLE-026-NP as new Item 4b, both as the appropriate analytical methods for TS 3.2.1, Heat Flux Hot Channel Factor.
    c. New Items 4a and 4 b are revised to add TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor, to the list of supported TS for these analytical methods.
    d. Item 5 is revised to list both WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary) prior to the title and applicable Addendum 1 and Addendum 2, deleting the reference to WCAP-12473-A (Non-Proprietary) at the end of the list of these applicable addenda.
    e. Item 5 is revised to add WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12472-NP-A (Non-Proprietary) Addendum 4, September 2012, to the list of analytical methods.
    f. Item 6 is revised to delete TS 3.3.1, Reactor Trip System (RTS) Instrumentation, from the list of supported TS for this analytical method.

UFSAR Changes:

The following licensing basis changes to the UFSAR are proposed:

1. Table 1.6-1 is revised as follows:

   a. A new entry is added under DCD Section Number 4.3 for Westinghouse Topical Report Number WCAP-12472-P-A (P) and WCAP-12472-NP-A as BEACON: Core Monitoring and Operations Support System, Addendum 4, September 2012.

   b. The entry for Westinghouse Topical Report Number WCAP-12945-P-A (P) and WCAP-14747, Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis, Revision 1, March 1998, under DCD Section Number 16.1 is deleted.

   c. The entry for Westinghouse Topical Report Number WCAP-12472-P-A (P) and WCAP-12473-A under DCD Section Number 16.1 is revised to add Addendum 2, March 2001.

   d. A new entry is added under DCD Section Number 16.1 for Westinghouse Topical Report Number WCAP-12472-P-A (P) and WCAP-12472-NP-A as BEACON: Core Monitoring and Operations Support System, Addendum 4, September 2012.

   e. A new entry is added under DCD Section Number 16.1 for Westinghouse Topical Report Number WCAP-16009-P-A (P) and WCAP-16009-NP-A as Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), Revision 0, January 2005.

   f. A new entry is added under DCD Section Number 16.1 for Westinghouse Topical Report Number APP-GW-GLE-026 (P) and APP-GW-GLE-026-NP as Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000, Revision 1, January 2009.

2. Subsection 4.3.5, Reference 4, Beard, C. L. and Morita, T., “BEACON: Core Monitoring and Operations Support System,” WCAP-12472-P-A (Proprietary) and WCAP-12473-A (Nonproprietary), August 1994; Addendum 1, May 1996; and Addendum 2, March 2001, is revised to add “; and WCAP-12472-P-A (Proprietary) and WCAP-12472-NP-A (Nonproprietary) Addendum 4, September 2012.”

## 2.2 Changes to COL Appendix A TS 5.7, High Radiation Area

TS 5.7 identifies controls which must be applied to high radiation areas in compliance with 10 CFR Part 20. A typographical error is identified in TS 5.7.2, High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation. TS 5.7.2(d)(1), 5.7.2(d)(2), and 5.7.2(d)(3) states that each individual group entering such an area shall possess either one of two different specific radiation devices or a self reading dosimeter, along with the specific requirements for each. TS 5.7.2(d)(4) states that if the options of TS 5.7.2(d)(1) and 5.7.2(d)(3) “are impractical or determined to be inconsistent with the “As

Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously ***displaces [emphasis added]*** radiation dose rates in the area" may be used. The word "displaces" is clearly a typographical error in this sentence. Therefore, it is proposed to change TS 5.7.2(d)(4) to reference "a radiation monitoring device that continuously ***displays [emphasis added]*** radiation dose rates in the area" as an editorial change to correct this typographical error.

**Description of any Changes to Current Licensing Basis Documents**

COL Appendix A TS Changes:

The following changes to the COL Appendix A TS are proposed:

1. TS 5.7.2 is revised to state: "In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area."

## 2.3 Technical Evaluation of Other Impacts

An impact review determined that these proposed changes would have no impact on the AP1000 plant Probabilistic Risk Assessment (PRA) presented in UFSAR Chapter 19, including the Fire PRA, results and insights [e.g., core damage frequency (CDF) and large release frequency (LRF)]. The proposed changes to the COL Appendix A TS for maintaining core operating limits and reactor trip setpoints within limits are administrative changes consistent with the requirements described elsewhere in the TS and in the UFSAR, and do not impact any initiating event and do not introduce any new failure modes or mechanisms. There are no physical modifications to any structure, system, or component (SSC) as described in the UFSAR. Therefore, there is no impact to or addition of any SSC that is considered to be AP1000 Design Reliability Assurance Program (D-RAP) risk-significant (DCD Tier 1 Table 3.7-1 and UFSAR Table 17.4-1). There is no interface with the diverse actuation system (DAS), and no change to the design functions of the DAS to provide diverse reactor protection system functions.

The proposed administrative changes to the COL Appendix A TS do not adversely affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. There are no physical modifications to any SSC as described in the UFSAR. Therefore, no safety-related SSC or function is adversely involved. The proposed changes do not involve nor interface with any SSC accident initiator or initiating sequence of events related to the accidents evaluated in the UFSAR. The proposed changes do not adversely affect the establishment of core operating limits and reactor trip setpoints, and are consistent with the requirements described elsewhere in the TS and in the UFSAR. The proposed changes do not result in any increase in probability of an analyzed accident occurring, and do not require a change in the analyses of normal operation and anticipated operational occurrences. The proposed changes do not affect the radiological source terms (i.e., amounts and types of radioactive materials released, their release rates and release durations) used in the accident analyses.

The proposed changes do not require a change to procedures or method of control that adversely affects the performance of any safety-related design function as described in the UFSAR. There are no physical modifications to any SSC function as described in the UFSAR, and the analytical methods described are approved for use consistent with the requirements described elsewhere in the TS and in the UFSAR for determining the associated core operating limits and reactor trip setpoints. The physical operational requirements of the plant, including as-installed inspections, testing, and maintenance requirements, as described in the UFSAR are not changed. Therefore, there are no changes to procedures or a method of control that adversely impact the licensing basis.

The proposed changes do not adversely interface with or adversely affect safety-related equipment or a fission product barrier. There are no physical modifications to any SSC as described in the UFSAR, and the described analytical methods are consistent with the requirements described elsewhere in the TS and in the UFSAR and comply with the regulatory requirements described in the UFSAR. The proposed changes do not result in a new failure mode, malfunction or sequence of events that could adversely affect a radioactive material barrier or safety-related equipment. The proposed changes do not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in significant fuel cladding failures.

The proposed changes do not adversely affect safety-related equipment or equipment whose failure could initiate an accident. There are no physical modifications to any SSC as described in the UFSAR that would interface with or adversely affect safety-related equipment or a radioactive material barrier. The proposed changes do not adversely affect any safety-related equipment, design code limit allowable value, safety-related function or design analysis, nor do they adversely affect any safety analysis input or result, or design/safety margin. Instead, the changes proposed preserve required design/safety margins.

The TS Safety Limits are not affected. The Limiting Safety System Settings, Limiting Control Settings, and Limiting Conditions for Operation requirements continue to be met by the proposed administrative changes to the COL Appendix A TS, and the requirements for the control of core operating limits and reactor trip setpoints are not affected by these changes that are consistent with the requirements described elsewhere in the TS and in the UFSAR, so that affected safety system functions are met and maintained operable.

There are no radiation zone changes or radiological access control changes required because of these proposed changes. The proposed change to TS 5.7 regarding controls that must be applied to high radiation areas in compliance with 10 CFR Part 20 is an administrative change to correct a typographical error. There are no physical modifications to any SSC as described in the UFSAR that may affect the radiation protection requirements, and thus there are no changes required to the radiation protection design features described in UFSAR Section 12.3.

There are no fire area changes required because of these proposed changes. The proposed changes do not require any changes to the fire protection analysis described in UFSAR Appendix 9A.

There is no change to the risk-significant designation of SSCs within the D-RAP as described in UFSAR Table 17.4-1.

The proposed changes do not affect the containment, control, channeling, monitoring, processing or releasing of radioactive and non-radioactive materials. No effluent release path is affected. The types and quantities of expected effluents are not changed. Therefore, radioactive or non-radioactive material effluents are not affected.

The proposed changes do not affect plant radiation zones, controls under 10 CFR Part 20, and expected amounts and types of radioactive materials, as the proposed change to TS 5.7 regarding controls that must be applied to high radiation areas in compliance with 10 CFR Part 20 is an administrative change to correct a typographical error, and does not result in any changes to these radiological conclusions as described in the UFSAR. Therefore, individual and cumulative radiation exposures do not change.

The proposed changes do not affect the results of the aircraft impact assessment described in UFSAR Subsection 19F.4.

The proposed changes have no adverse impact on the emergency plan or the physical security plan implementation, because there are no changes to physical access to credited equipment inside the Nuclear Island (including containment or the auxiliary building) and no adverse impact to plant personnel's ability to respond to any plant operations or security event.

## 2.4 Summary

The proposed changes would revise COL Appendix A TS information, and associated UFSAR information, concerning the analytical methods for maintaining core operating limits and reactor trip setpoints within limits, and are administrative changes consistent with the requirements as described elsewhere in the TS and in the UFSAR. The proposed changes do not adversely affect the design functions of any SSC as described in the UFSAR.

The proposed changes do not adversely affect any safety-related equipment or function, design function, radioactive material barrier, or safety analysis.

# 3. TECHNICAL EVALUATION (Included in Section 2)

# 4. REGULATORY EVALUATION

## 4.1 Applicable Regulatory Requirements/Criteria

10 CFR 52.98(c) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. The proposed changes involve a change to COL Appendix A TS information. Therefore, NRC approval is required prior to making the plant-specific proposed changes in this license amendment request.

10 CFR 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the TS, or requires a license amendment under paragraphs B.5.b or B.5.c of

the section. The proposed changes, which include changes to the UFSAR, involve a revision to COL Appendix A TS information. Therefore, NRC approval is required for the Tier 2 and the TS changes.

10 CFR 52, Appendix D, VIII.C.6 states that after issuance of a license, “Changes to the plant-specific TS (Technical Specifications) will be treated as license amendments under 10 CFR 50.90.” 10 CFR 50.90 addresses the applications for amendments of licenses, construction permits and early site permits. As discussed above, changes to TS are requested. Therefore, NRC approval is required for these TS changes.

10 CFR 50.36 Technical specifications. - (c) *Technical specifications will include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings. (2) Limiting conditions for operation. (3) Surveillance Requirements.* The safety limits are not affected. In addition, except where justified by this license amendment request, the limiting safety system settings, limiting control settings, and limiting conditions for operation requirements continue to be met with the proposed administrative changes involving reactor trip setpoints, and involving reactivity control, RCS, and power distribution core operating limits requirements. The existing reactor trip system instrumentation safety functions, and the existing reactivity control, RCS, and power distribution core operating limits requirements, are met by these proposed administrative changes, and there are no proposed changes to the existing COL Appendix A TS LCO, Action, and SR requirements.

10 CFR 20.1101(b) requires that the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). The proposed change to TS 5.7 regarding controls that must be applied to high radiation areas in compliance with 10 CFR Part 20 is an administrative change to correct a typographical error. Therefore, there are no changes necessary to the procedures and engineering controls to achieve occupational doses and doses to members of the public ALARA.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 10 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. Although the proposed changes involve reactor trip setpoints, and involve reactivity control, RCS, and power distribution core operating limits requirements, the changes are administrative and do not involve physical modifications or addition of SSCs, and do not change the analyses of any condition of normal operation, including the effects of anticipated operational occurrences. Therefore, the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, are not changed, and the fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects. Therefore, the proposed changes comply with the requirements of GDC 10.

10 CFR Part 50, Appendix A, GDC 11 requires that the reactor core and associated coolant systems be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. The proposed administrative changes involving reactor trip setpoints, and involving

reactivity control, RCS, and power distribution core operating limits requirements, maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that the acceptance criteria for events resulting in positive reactivity insertion and reactivity feedback effects are met. Therefore, the proposed changes comply with the requirements of GDC 11.

10 CFR Part 50, Appendix A, GDC 12 requires a core design to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible. The proposed administrative changes involving reactor trip setpoints, and involving reactivity control, RCS, and power distribution core operating limits requirements, prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects. Therefore, the proposed changes comply with the requirements of GDC 12.

10 CFR Part 50, Appendix A, GDC 15 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 normal operation, including anticipated operational occurrences. The proposed administrative changes involving reactor trip setpoints, and involving reactivity control, RCS, and power distribution core operating limits requirements, maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that RCS design limits are not exceeded. Therefore, the proposed changes comply with the requirements of GDC 15.

10 CFR Part 50, Appendix A, GDC 20 requires that a protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The proposed administrative changes involving maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for anticipated operational occurrences and postulated accidents. The proposed changes do not involve physical modifications or addition of SSCs. Therefore, the proposed changes comply with the requirements of GDC 20.

10 CFR Part 50, Appendix A, GDC 25 requires that a protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of the control rods. The proposed administrative changes involving reactor trip setpoints, and involving reactivity control, RCS, and power distribution core operating limits requirements, maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for any single malfunction of the reactivity control systems. The proposed changes do not involve physical modifications or addition of SSCs. Therefore, the proposed changes comply with the requirements of GDC 25.

10 CFR Part 50, Appendix A, GDC 28 requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. The proposed administrative changes involving reactor trip setpoints, and involving reactivity control, RCS, and power distribution core operating limits requirements, maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that the safety functions of the reactivity control systems are met. The proposed changes do not involve physical modifications or addition of SSCs. Therefore, the proposed changes comply with the requirements of GDC 28.

## 4.2 Precedent

None.

## 4.3 Significant Hazards Consideration

The proposed changes would revise the Combined License (COL) regarding the analytical methods approved for maintaining core operating limits and reactor trip setpoints within limits, and to correct a typographical error regarding controls for high radiation areas, and are administrative and editorial changes consistent with the requirements described elsewhere in the COL Appendix A Technical Specifications (TS) and in the Updated Final Safety Analysis Report (UFSAR).

The requested amendment proposes changes to COL Appendix A TS and UFSAR Tier 2 information.

An evaluation to determine whether a significant hazards consideration is involved with the requested amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

### 4.3.1 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed changes are administrative and editorial changes consistent with the requirements described elsewhere in the TS and in the UFSAR, and do not adversely affect the operation of any systems or equipment that initiate an analyzed accident or alter any structures, systems, and components (SSCs) accident initiator or initiating sequence of events. The proposed changes to the analytical methods approved for maintaining core operating limits do not result in any increase in probability of an analyzed accident occurring, and prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects, and so that the consequences of postulated accidents are not changed. The proposed changes do not adversely affect the ability of the automatic reactor trips to perform the required safety function to trip the reactor when necessary to protect fuel design limits, and do not adversely affect the probability of inadvertent operation or failure of the automatic reactor trips.

Therefore, the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed changes are administrative and editorial changes consistent with the requirements described elsewhere in the TS and in the UFSAR, and do not affect the operation of any systems or equipment that may initiate a new or different kind of accident, or alter any SSC such that a new accident initiator or initiating sequence of events is created. The proposed changes to the analytical methods approved for maintaining core operating limits do not result in any increase in probability of an analyzed accident occurring, and prevent power oscillations and maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that fuel design limits are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects, and so that the consequences of postulated accidents are not changed. The proposed changes do not adversely affect the ability of the automatic reactor trips to perform the required safety function to trip the reactor when necessary to protect fuel design limits, and do not adversely affect the probability of inadvertent operation or failure of the automatic reactor trips.

These proposed changes do not adversely affect any other SSC design functions or methods of operation in a manner that results in a new failure mode, malfunction, or sequence of events that affect safety-related or nonsafety-related equipment. Therefore, this activity does not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that results in significant fuel cladding failures.

Therefore, the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

**4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No

The proposed changes are administrative and editorial changes consistent with the requirements described elsewhere in the TS and in the UFSAR, and maintain existing safety margins through continued application of the existing requirements of the UFSAR. The proposed changes maintain the initial conditions and operating limits required by the accident analysis, and the analyses of normal operation and anticipated operational occurrences, so that the existing fuel design limits specified in the UFSAR are not exceeded for events resulting in positive reactivity insertion and reactivity feedback effects, and so that the consequences of postulated accidents are not changed. Therefore, the proposed changes satisfy the same safety functions in accordance with the same requirements as stated in the

UFSAR. These changes do not adversely affect any design code, function, design analysis, safety analysis input or result, or design/safety margin.

No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes, and no margin of safety is reduced. Therefore, the requested amendment does not involve a significant reduction in a margin of safety.

## 4.4 Conclusions

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. Therefore, it is concluded that the requested amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

# 5. ENVIRONMENTAL CONSIDERATIONS

The requested amendment requires changes to the Combined License (COL) regarding the analytical methods approved for maintaining core operating limits and reactor trip setpoints within limits, and to correct a typographical error regarding controls for high radiation areas, and are administrative and editorial changes consistent with the requirements described elsewhere in the COL Appendix A Technical Specifications (TS) and in the Updated Final Safety Analysis Report (UFSAR).

The requested amendment proposes changes to COL Appendix A TS and UFSAR Tier 2 information.

A review has determined that the anticipated effects on facility construction and operation following implementation of the requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

*(i) There is no significant hazards consideration.*

As documented in Section 4.3, Significant Hazards Consideration, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment." The Significant Hazards Consideration determined that (1) the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the requested amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the requested amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

*(ii)* *There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.*

The proposed changes are unrelated to any aspect of plant construction or operation that would introduce any change to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents), or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the design functions or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the requested amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

*(iii)* *There is no significant increase in individual or cumulative occupational radiation exposure.*

The proposed changes do not adversely affect walls, floors, or other structures that provide shielding. Plant radiation zones are not affected, and there are no changes to the controls required under 10 CFR Part 20 that could result in a significant increase in occupational radiation exposure. Therefore, the requested amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the requested amendment, it has been determined that anticipated construction and operational impacts of the requested amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the requested amendment meets the eligibility criteria 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 requested amendment.

## 6. REFERENCES

6.1. WCAP-16361-P-A (Westinghouse Proprietary) and WCAP-16361-NP-A (Non-Proprietary), "Westinghouse Setpoint Methodology for Protection Systems – AP1000," Revision 1, February 2011 (NRC Accession Number ML110601158).

6.2. APP-GW-GLR-137, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," Revision 1 (NRC Accession Number ML110620129).

6.3. WCAP-9272-P-A (Westinghouse Proprietary) and WCAP-9273-NP-A (Non-Proprietary), "Westinghouse Reload Safety Evaluation Methodology," July 1985.

6.4. WCAP-16009-P-A (Westinghouse Proprietary) and WCAP-16009-NP-A (Non-Proprietary), "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005 (NRC Accession Number ML050910157).

6.5. WCAP-17524-P-A, "AP1000 Core Reference Report," Revision 1 (Westinghouse Proprietary) (NRC Accession Number ML15180A174).

6.6. WCAP-12945-P-A (Westinghouse Proprietary) and WCAP-14747 (Non-Proprietary), Volumes 1-5, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," Revision 2, March 1998 (NRC Accession Number 9804070248).

6.7. APP-GW-GLE-026 (Westinghouse Proprietary) and APP-GW-GLE-026-NP (Non-Proprietary), "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000", Revision 1, February 2009 (NRC Accession Number ML090410367).

6.8. WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary), "BEACON: Core Monitoring and Operations Support System," August 1994 (NRC Accession Number 9409280012); Addendum 1, May 1996 (NRC Accession Number ML003678340); and Addendum 2, March 2001 (NRC Accession Number ML021270086).

6.9. WCAP-16045-P-A (Westinghouse Proprietary) and WCAP-16045-NP-A (Non-Proprietary), "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

6.10. WCAP-16045-P-A (Westinghouse Proprietary) and WCAP-16045-NP-A (Non-Proprietary) Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.

6.11. WCAP-12472-P-A (Westinghouse Proprietary) Addendum 4 (NRC Accession Number ML12270A385) and WCAP-12472-NP-A (Non-Proprietary) Addendum 4 (NRC Accession Number ML12270A386), "BEACON: Core Monitoring and Operations Support System," September 2012,

**Southern Nuclear Operating Company**

**ND-18-0057**

**Enclosure 2**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Proposed Changes to the Licensing Basis Documents**

**(LAR-18-006)**

**Additions identified by blue underlined text.**
**Deletions identified by red strikethrough of text.**
*** * * indicates omitted existing text that is not shown.**

(This Enclosure consists of 7 pages, including this cover page.)

**COL Appendix A, Technical Specification (TS) 5.6.3, CORE OPERATING LIMITS REPORT (COLR), is revised as follows:**

1. TS 5.6.3.a is revised as follows:

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

* * *

~~3.3.1, "Reactor Trip System (RTS) Instrumentation";~~

* * *

2. TS 5.6.3.b is revised as follows:

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Westinghouse Proprietary) and WCAP-9273-NP-A (Non-Proprietary).

(Methodology for Specifications 3.1.1 - Shutdown Margin (SDM), 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limits, 3.1.6 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - AXIAL FLUX DIFFERENCE, 3.4.1 - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, and 3.9.1 - Boron Concentration.)

* * *

4a. ~~WCAP-12945-P-A, Volumes 1-5, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," Revision 2, March 1998 (Westinghouse Proprietary) and WCAP-14747 (Non-Proprietary)~~WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005 (Westinghouse Proprietary) and WCAP-16009-NP-A (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

4b. APP-GW-GLE-026, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000", Revision 1, February 2009 (Westinghouse Proprietary) and APP-GW-GLE-026-NP (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

5. WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary), "BEACON Core Monitoring and Operations Support System," August 1994, Addendum 1, May 1996~~ (Westinghouse Proprietary)~~, and Addendum 2, March 2001~~, (Westinghouse Proprietary), and WCAP-12473-A (Non-Proprietary)~~; and WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12472-NP-A (Non-Proprietary) Addendum 4, September 2012.

   (Methodology for Specification 3.2.5 - OPDMS - Monitored Parameters.)

6. APP-GW-GLR-137, Revision 1, "Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips," Westinghouse Electric Company LLC.

   (Methodology for Specification 2.1.1 - Reactor Core Safety Limits~~, and 3.3.1 - Reactor Trip System (RTS) Instrumentation~~.)

* * *

**COL Appendix A, TS 5.7, High Radiation Area, is revised as follows:**

1. TS 5.7.2 is revised as follows:

    5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

    * * *

    d. Each individual group entering such an area shall possess:

    * * *

    4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously ~~displaces~~displays radiation dose rates in the area.

    * * *

**UFSAR Tier 2 Table 1.6-1 (Sheet 6 of 21), Material Referenced, is revised as follows:**

**Table 1.6-1 (Sheet 6 of 21)**
**Material Referenced**

| DCD Section Number | Westinghouse Topical Report Number | Title |
|---|---|---|
| * * * | * * * | * * * |
| 4.3 | * * * | * * * |
| | WCAP-12472-P-A (P)<br>WCAP-12473-A | BEACON: Core Monitoring and Operations Support System, August 1994; Addendum 1, May 1996; Addendum 2, March 2001 |
| | WCAP-12472-P-A (P)<br>WCAP-12472-NP-A | BEACON: Core Monitoring and Operations Support System, Addendum 4, September 2012 |
| | * * * | * * * |
| * * * | * * * | * * * |

**UFSAR Tier 2 Table 1.6-1 (Sheet 18 of 21), Material Referenced, is revised as follows:**

**Table 1.6-1 (Sheet 18 of 21)**
**Material Referenced**

| DCD Section Number | Westinghouse Topical Report Number | Title |
|---|---|---|
| * * * | * * * | * * * |
| 16.1 | * * * | * * * |
| | ~~WCAP-12945-P-A (P)~~<br>~~WCAP-14747~~ | ~~Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis, Revision 1, March 1998~~ |
| | WCAP-12472-P-A (P)<br>WCAP-12473-A | BEACON Core Monitoring and Operations Support System, August 1994~~, and~~; Addendum 1, May 1996; and Addendum 2, March 2001 |
| | WCAP-12472-P-A (P)<br>WCAP-12472-NP-A | BEACON: Core Monitoring and Operations Support System, Addendum 4, September 2012 |
| | * * * | * * * |
| | WCAP-16779 | AP1000 Overpressure Protection Report, April 2007 |
| | WCAP-16009-P-A (P)<br>WCAP-16009-NP-A | Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), Revision 0, January 2005 |
| | APP-GW-GLE-026 (P)<br>APP-GW-GLE-026-NP | Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000, Revision 1, January 2009 |
| | * * * | * * * |
| * * * | * * * | * * * |

**UFSAR Tier 2 Subsection 4.3.5, References, is revised as follows:**

## 4.3.5 References

* * *

4. Beard, C. L. and Morita, T., "BEACON: Core Monitoring and Operations Support System," WCAP-12472-P-A (Proprietary) and WCAP-12473-A (Nonproprietary), August 1994; Addendum 1, May 1996; and Addendum 2, March 2001; and WCAP-12472-P-A (Proprietary) and WCAP-12472-NP-A (Nonproprietary) Addendum 4, September 2012.

* * *