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SCE&G®
A SCANA COMPANY

August 26, 2010
NND-10-0319

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

ATTN: Document Control Desk

Subject: Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 Combined License Application (COLA) - Docket Numbers 52-027 and 52-028 Submittal of Roadmap of Changes for the August 25, 2010 Submittal of the COLA.

References:

1. Letter from Ronald B. Clary (SCE&G) to Document Control Desk (NRC), August 25, 2010 Submittal of the Combined License Application for V.C. Summer Nuclear Station Units 2 and 3, Including Revision 3 of the Final Safety Analysis Report (FSAR).

By letter dated March 27, 2008, South Carolina Electric & Gas Company (SCE&G) submitted a combined license application (COLA) for two Westinghouse AP1000 units, designated V.C. Summer Nuclear Station (VCSNS) Units 2 and 3, to be located at the existing VCSNS site in Fairfield County, South Carolina. By letter dated August 25, 2010, SCE&G submitted an update to the VCSNS Units 2 and 3 COLA. Enclosed is a 'roadmap' of changes included in the August 25, 2010 update, which is being provided as a reviewer aid. Also provided is an explanation of the roadmap column headings.

Should you have any questions, please contact me by telephone at (803) 345-4191, or by email at apaglia@scana.com.

Very truly yours,

Alfred M. Paglia
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JEF/AMP/jf

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NRO

Enclosure – COLA August 25, 2010 Submittal Roadmap

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NND-10-0319

Enclosure

VCSNS Units 2 and 3 COL Application

August 25, 2010 Submittal Roadmap

(55 Pages Including Coversheet and Column Index Page)

VCSNS Units 2 & 3 COL Application August 25, 2010 Submittal Roadmap Format Explanation
(by columns)

Column Label	Explanation
Change ID #	Unique internal identifier for tracking purposes
COLA REP	Identifies the change as STD (standard) or VCSNS (VCS) specific
COLA Part REP	Part 1 (PT01) through Part 15 (PT15)
Chapter REP	FSAR Chapter
Section / Page REP	Identifies specific location of the change in the COLA. Page numbers, if identified, are specific to the document that was revised, i.e., Revision 3.
Complete Change Description	Short description of the change
Basis for Change	The source of the change

NuStart's COLA Tracking Management (CTM) : COLA Changes | Cola Rev3 Roadmp

AUG-26-2010 1:55 PM

Cola Rev3 Roadmp

VC Summer is 1 AND ...

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
PT01 - *(empty)*						**1 COLA Change**
7796	VCS	PT01		Section 1.0 / Page 1	In Part 1, 1.0 Introduction, add the following entry in the listing of Application Parts: Part 15 - COLA Enclosure 5 - Cyber Security Plan	Inclusion of New Part 15 - Cyber Security Plan with COLA Submittal
PT02 - FSAR01						**38 COLA Changes**
7514	VCS	PT02	FSAR01	01.04	Revise COLA Part 2, FSAR Section 1.4 as shown in SCE&G Voluntary Letter Number NND-10-0247, dated June 29, 2010.	Verbal request for clarification from NRC staff provided in SCE&G Letter NND-10-0247 dated June 29, 2010.
7701	VCS	PT02	FSAR01	01.06.T / T1.6-201	COLA Part 2, FSAR Chapter 1, Table 1.6-201, will be revised to update Cyber Security Plan revision and date consistent with June 2010 submittal of Revision 0, to read: Author/Report Number(a) — Title — Revision — FSAR Section — Document Transmittal — ADAMS Accession Number Cyber Security Plan — VCSNS 2 and 3 Cyber Security Plan — 0 — 13.6 — June 2010 — Not Applicable (SUNSI)	VEGP-LTR-051 S1 response (SNC Ltr ND-10-1178 dated 20100614)
7722	VCS	PT02	FSAR01	01.06.T / T1.6-201	Revise FSAR TAble 1.6-201 to add a horizontal separator line after the "NEI 07-02A" entry. Add left margin annotation VCS SUP 1.6-2.	Editorial
7549	VCS	PT02	FSAR01	01.08.T / T1.8-201	Revise VCSNS FSAR Subsection 2.2.2.2.1.1 to add left margin annotation VCS COL 6.4-1.	SCE&G Letter NND-10-0248, Supplement to SCE&G Letter NND-10-0235 endorsement of BLN RAI LTR 169 response to RAI 06.04-008.
7560	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 02.05-17	1. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, will be revised to add the following new line item for the "Waterproofing System" to read: 2.5-17 Waterproofing System 2.5.4.6.12 2.5.7 A	DCD Rev 18, VEGP-VOL-CH02 re waterproofing in response to VEGP-COL-02.05-017 item 1, (SNC Ltr ND-10-1281) RAI 3.7.1-2 (Letter 53 S1 Response per SCE&G Letter NND-

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
						10-0261)
7669	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 02.05-17	1. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, will be revised to add the following new line item for the "Waterproofing System" to read: 2.5-17 Waterproofing System 2.5.4.6.12 2.5.4.14 3.8.5.1 A	Editorial correction to Change ID 7560 and RAI 3.7.1-2 (Letter 53 S1 Response per SCE&G Letter NND-10-0261) to reference correct subsection
7717	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 03.06-01	COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, COL Item 3.6-1 Pipe Break Hazards Analysis will be revised to add FSAR Section 14.3.3.1 to the list of FSAR Sections.	Editorial
7720	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 03.09-05	COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, COL Item 3.9-5 Surge Line Thermal Monitoring will be revised to add FSAR Section 14.2.9.2.22 to the list of FSAR Sections.	Editorial- Section 14.2.9.2.22 provides additional surge line thermal monitoring information.
7719	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 03.09-07	COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, COL Item 3.9-7 As-Designed Piping Analysis will be revised to add FSAR Section 14.3.3.2 to the list of FSAR Sections.	Editorial
7324	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 07.01-01	1. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, will be revised to include the following new line item to address COL Information Item 7.1-1: 7.1-1 Setpoint Calculations for Protective Functions 7.1.6.1 7.1.6.1 B	DCD Rev 18, VEGP-VOL-CH07 response to 07.01-001 item 1 SNC Ltr ND-10-1118 VEGP-VOL-CH07 S1 response to 07.01-001 item 1 SNC Ltr ND-10-1266
7325	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 07.05-01	1. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, will be revised to include the following new line item to address COL Information Item 7.5-1: 7.5-1 Post Accident Monitoring System 7.5.5 7.5.2, 7.5.3.5, 7.5.5 A	DCD Rev 18, VEGP-VOL-CH07 S1 item 1, SNC Ltr ND-10-1266
7326	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 11.5-1,2,3	Revise COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, items 11.5-1, 11.5-2, and 11.5-3 to renumber each DCD and FSAR reference from: 11.5.7 To read: 11.5.8	DCD Rev 18, Based on WEC letter DCP/NRC2492 dated 20090522
7721	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 13.06-1	COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, COL Item 13.6-1, Security, will be revised to add FSAR Section 14.3.2.3.2 to the list of FSAR Sections.	Editorial
7328	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 15.0-1	1. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, will be revised to include a new line item for COL item 15.0-1 as follows: 15.0-1 Documentation of Plant Calorimetric Uncertainty Methodology 15.0.15 15.0.15.1 H	DCD Rev 18, Based on WEC letter DCP/NRC2461 dated 20090506 COL-SER-OI-Ch15 S1 response to OI 15.00-01 item 1 (SNC Ltr ND-10-1018)
7728	STD,VCS	PT02	FSAR01	01.08.T / T1.8-202 15.0-1	1. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-202, will be revised to switch the DCD and FSAR references for new line item for COL item 15.0-1 from:	DCD Rev 18, Based on WEC letter DCP/NRC2461 dated

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					15.0-1 Documentation of Plant Calorimetric Uncertainty Methodology 15.0.15 15.0.15.1 H To read: 15.0-1 Documentation of Plant Calorimetric Uncertainty Methodology 15.0.15.1 15.0.15 15.0.3.2 H	20090506 Editorial for consistency with DCD and with FSAR Chapter 15 changes identified in COL-SER-OI-Ch15 S1 response to OI 15.00-01 item 1 SNC Ltr ND-10-1018
7602	STD,VCS	PT02	FSAR01	01.08.T / T1.8-203 07.04	2. COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-203, will be revised to include the following plant interface item: Item No. Interface Interface Type Matching Interface Item Section or Subsection(1) 7.4 Post Accident Monitoring System NNS Combined License Applicant Coordination 7.5.5	DCD Rev 18, VEGP-VOL-CH07 S1 item 2, SNC Ltr ND-10-1266
7603	STD,VCS	PT02	FSAR01	01.08.T / T1.8-203 07.04	COLA Part 2, FSAR Chapter 1, Section 1.8, Table 1.8-203, item 7.4, Matching Interface Item, will be revised from: Combined License Applicant Coordination To read: Combined License applicant coordination	DCD Rev 18, Editorial revision to VEGP-VOL-CH07 S1 item 2, SNC Ltr ND-10-1266
7332	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.008	7. COLA Part 2, FSAR Chapter 1, Section 1.9, Table 1.9-201, for RG 1.8, will be revised to add an additional cross-reference of: 17.5 (QAPD, IV)	COL-SER-OI-Ch17 S1 response to OI 17.01-001 item 7 (SNC Letter ND-09-2082)
7546	VCS	PT02	FSAR01	01.09.T / T1.9-201 1.028	8. COLA Part 2, FSAR Chapter 1, Section 1.9, Table 1.9-201, for RG 1.28, will be revised from: 1.28 Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985) Not referenced; see Appendix 1AA To read: 1.28 Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985) 17.5 (QAPD, II, 17.1) 17.5 (QAPD, IV)	COL-SER-OI-Ch17 S1 response to OI 17.01-001 item 8 (SNC Letter ND-09-2082) While this change is listed as plant specific in the letter, it is a standard section and applies to the VCSNS application.
7333	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.033	1. COLA Part 2, FSAR Chapter 1, Section 1.9, Table 1.9-201, for RG 1.33, will be revised from: 1.33 Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978) 16 (TS 5.4.1) To read: 1.33 Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978) 16 (TS 5.4.1) 17.5 (QAPD, IV)	COL-SER-OI-Ch17 S1 response to OI 17.01-001 item 1 (SNC Letter ND-09-2082)
7334	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201	9. COLA Part 2, FSAR Chapter 1, Section 1.9, Table 1.9-201, for RG 1.37, will be revised to add an additional	COL-SER-OI-Ch17 S1

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
				1.037	cross-reference of: 17.5 (QAPD, II, 13.2)	response to OI 17.01-001 item 9 (SNC Letter ND-09-2082)
7335	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.052	COLA Part 2, FSAR Chapter 1, Section 1.9, Table 1.9-201, will be revised to add new item for RG 1.52 to read: 1.52 Design, Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001) 16 (TS 3.7.6)	DCD Rev 18, Based on WEC letter DCP/NRC2457 dated 20090504
7336	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.053	COLA Part 2, FSAR Chapter 1, Table 1.9-201, title for Regulatory Guide 1.53 will be revised from: "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems" To read: "Application of the Single-Failure Criterion to Safety Systems"	Editorial
7337	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.135	Revise COLA Part 2, Chapter 1, Section 1.9, Table 1.9-201, for Regulatory Guide 1.135, from a listing in the FSAR Chapter, Section, or Subsection column of: Not referenced; see Appendix 1AA To read: DCD discussion only; see DCD Table 1.9-1	RG is referenced in DCD but not in COLA
6644	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.078	For RG 1.78, add cross reference to Table 19.58-201	Consistency
7330	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.078	1. Revise COLA Part 2, Chapter 1, Section 1.9, Table 1.9-201, to include the following additional cross-reference listing in the FSAR Chapter, Section, or Subsection column for Regulatory Guide 1.78: 6.4.3	BLN RAI LTR 168 response to RAI 06.04-007 item 1
7338	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.192	(4) Revise Subsection 1.9, Table 1.9-201, for Regulatory Guide 1.192, from a listing in the FSAR Chapter, Section, or Subsection column of: Not referenced; see Appendix 1 AA To read: 3.9.6.3	COL-SER-OI-Ch03 S7 response to OI 03.09-002(b) item 4 (SNC Ltr ND-10-0949)
7331	STD,VCS	PT02	FSAR01	01.09.T / T1.9-201 1.196	2. Revise COLA Part 2, Chapter 1, Section 1.9, Table 1.9-201, for Regulatory Guide 1.196, from a listing in the FSAR Chapter, Section, or Subsection column of: Not referenced; see Appendix 1AA To read: 6.4.3	BLN RAI LTR 168 response to RAI 06.04-007 item 2
7555	VCS	PT02	FSAR01	01.09.T / T1.9-202 Sh 2	Revise VCSNS FSAR Table 1.9-202 (Sheet 2 of 20), Criteria Section 2.5.2 as shown in SCE&G Letter NND-10-0256.	Response to Request for Additional Information No. 4837 Revision 2 received June 25, 2010 per SCE&G

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
						Letter NND-10-0256.
7643	STD,VCS	PT02	FSAR01	01.09.T / T1.9-204 BU88-11	1. COLA Part 2, FSAR Chapter 1, Table 1.9-204, will be revised to include a new Bulletin line item, to read: 88-11 Pressurizer Surge Line Thermal Stratification 3.9.3.1.2	VEGP-RAI-LTR 057 response to RAI 03.12-002 item 1, SNC Ltr ND-10-1263
7339	STD,VCS	PT02	FSAR01	01.09.T / T1.9-204 GL85-05	COLA Part 2, FSAR Chapter 1, Section 1.9, Table 1.9-204, will be revised to restore the line item for Generic Letter 85-05 as follows: 85-05 Inadvertent Boron Dilution Events (1/85) 13.5	COL-SER-OI-Ch15 response to OI 15.04-001 (SNC Ltr ND-10-0004)
7340	STD,VCS	PT02	FSAR01	01AA 1.033	2. COLA Part 2, FSAR Chapter 1, Appendix 1AA, for RG 1.33, will be revised from: Regulatory Guide 1.33, Rev. 2, 2/78 – Quality Assurance Program Requirements (Operation) General Exception Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards. To read: Regulatory Guide 1.33, Rev. 2, 2/78 – Quality Assurance Program Requirements (Operation) C.1 & C.3 Conforms C.2 Clarification See separate conformance statement for each identified Regulatory Guide. C.3 – C.5 Conforms	COL-SER-OI-Ch17 S1 response to OI 17.01-001 item 2 (SNC Letter ND-09-2082)
7341	STD,VCS	PT02	FSAR01	01AA 1.033	COLA Part 2, FSAR Chapter 1, Appendix 1AA, for RG 1.33, will be revised to remove the "& C.3" from the first "Conforms"	C.3 is also addressed in "C.3 - C.5" conformance statement
7342	STD,VCS	PT02	FSAR01	01AA 1.052	COLA Part 2, FSAR Chapter 1, Appendix 1AA, will be revised to add new item for RG 1.52 to read: Regulatory Guide 1.52, Rev. 3, 6/01 - Design, Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants Conformance with the design and operational aspects is as stated in the DCD.	DCD Rev 18, Based on WEC letter DCP/NRC2457 dated 20090504
7343	STD,VCS	PT02	FSAR01	01AA 1.083	COLA Part 2, FSAR Chapter 1, Appendix 1AA, for RG 1.83, will be revised from: Conformance of the design aspects is as stated in the DCD. Conformance of the programmatic and/or operational aspects is documented below. General Exception Steam generator tube surveillance is in accordance with Nuclear Energy Institute (NEI) 97-06. This guidance has been endorsed by NRC as an acceptable program basis. To read:	This RG withdrawn by NRC.

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					Conformance of the design aspects is as stated in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 58324, 11/12/2009).	
7344	STD,VCS	PT02	FSAR01	01AA 1.084	COLA Part 2, FSAR Chapter 1, Appendix 1AA, RG 1.84, will be revised from: Conformance with Revision 31 of the Regulatory Guide is as stated in the DCD. To read: Conformance with Revision 32 of the Regulatory Guide is as stated in the DCD.	DCD Rev 18, Based on WEC letter DCP/NRC2533 dated 20090617
7345	STD,VCS	PT02	FSAR01	01AA 1.133	COLA Part 2, FSAR Chapter 1, Appendix 1AA, Regulatory Guide 1.133, position C.6, will be revised to read: C.6 Exception Regulatory Guide 1.16 has been withdrawn. Event reporting is performed in accordance with 10 CFR 50.72 and 50.73 utilizing the guidance of NUREG-1022	COL-SER-CI-Ch04 response to CI 04.04-001 SNC Letter ND-10-0006
7210	VCS	PT02	FSAR01	01AA RG 1.028	Revise Regulatory Guide 1.28 with, LMA VCS COL 1.9-1 from: --- Regulatory Guide 1.28, Rev. 0, 6/72 - Quality Assurance Program Requirements (Design and Construction) Design, procurement, and construction activities associated with VCSNS Units 2 and 3 that may occur before the COL is issued will be conducted in accordance with the existing SCE&G Unit 1 QA Program requirements (Regulatory Guide 1.28, Rev. 0). Design and construction activities that occur following COL issuance will be conducted in accordance with the QAPD submitted as Part 13 of the application and is the QAPD that was evaluated and discussed in Table 1.9-202 for conformance to SRP 17.1. --- to read: --- Regulatory Guide 1.28, Rev. 0, 6/72 - Quality Assurance Program Requirements (Design and Construction) Design, procurement, and construction activities associated with VCSNS Units 2 and 3 that may occur before the COL is issued will be conducted in accordance with the existing SCE&G Unit 1 QA Program requirements (Regulatory Guide 1.28, Rev. 0). Design and construction activities that occur following COL issuance will be conducted in accordance with the QAPD submitted as Part 13 of the application and is the QAPD that was evaluated and discussed in Table 1.9-202 for conformance to SRP 17.1. Regulatory Guide 1.28, Rev. 3, 8/85 - Quality Assurance Program Requirements (Design and Construction) Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below. General Exception Quality assurance requirements utilize the more recently NRC endorsed NQA-1-1994 in lieu of the identified outdated standards. ---	RAI 17.5-6 S1 (Letter 012 Supplemental Response 1 per NND-09-0295)
PT02 - FSAR02						**27 COLA Changes**

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
7511	VCS	PT02	FSAR02	02.00.T / T 2.0-201	In Table 2.0-201, revise Plant Specific values for Snow/Ice Loads from: "100-year return period ground-level snowpack of 12.2 pounds per square foot" to read: "100-year return period ground-level snowfall of 12.4 pounds per square foot."	RAI 02.03.01-9 (NND-10-0230)
7564	VCS	PT02	FSAR02	02.00.T / T 2.0-201	Revise note (k) in Table 2.0-201 to read as follows: (k) Justification for the exemption and departure for Maximum Safety Wet Bulb Temperature (Noncoincident) can be located in the following FSAR Subsections: 2.3.1.5, 5.4.7.1.2.3, 6.2.1.1.3, 6.2.2.3, 6.4, 9.1.3.1.3.1, 9.2.2.1.2.1, and 9.2.7.2.4.	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7579	VCS	PT02	FSAR02	02.00.T / T2.0-201	COLA Part 2, FSAR Chapter 2, Section 2.0, Table 2.0-201 , third column , under the "Seismic" related Plant Specific Value "Fault Displacement Potential," will be revised from "Negligible" to read: There are no capable faults within 25 miles of the VCSNS site vicinity. Therefore, there is no fault displacement potential within this area.	DCD Rev 18, Based on WEC letter DCP/NRC2897 dated 20100528 SCE&G Letter NND-10-0260
7705	VCS	PT02	FSAR02	02.00.T / T2.0-201	Revise note (k) in Table 2.0-201 to read as follows: (k) Justification for the exemption and departure for Maximum Safety Wet Bulb Temperature (Noncoincident) can be located in the following FSAR Subsections: 2.3.1.5, 5.4.7.1, 6.2.1.1.3, 6.2.2.3, 6.4, 6.4.1.1, 9.1.3.1.3.1, 9.2.2.1, and 9.2.7.2.4.	Editorial Correction of Subsection Pointers provided in Change ID 7564 and RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7580	VCS	PT02	FSAR02	02.00.T / T2.0-201	COLA Part 2, FSAR Chapter 2, Section 2.0, Table 2.0-201 , third column, under the "Soil" related Plant Specific Value "Liquefaction Potential" will be revised from Negligible to read: Units 2 and 3 are founded on hard rock. The seismic Category II Annex Building and 1st Bay of the Turbine Building, along with other portions of the Turbine Building and Radwaste Building, are founded on engineered structural fill on top of the hard rock. This fill does not have the potential to liquefy.	DCD Rev 18, Based on WEC letter DCP/NRC2897 dated 20100528 SCE&G Letter NND-10-0260
7581	STD,VCS	PT02	FSAR02	02.00.T / T2.0-201	COLA Part 2, FSAR Chapter 2, Section 2.0, Table 2.0-201, Sheet 4 of 5, revise note (f) from: (f) Sites that fall within the hard rock high frequency GMRS given in DCD Figures 3I.1-1 and 3I.1-2 are acceptable. To read: (f) Sites that fall within the hard rock high frequency envelope response spectra given in DCD Figures 3I.1-1 and 3I.1-2 and satisfy the limitation on shear wave velocity in DCD Subsection 2.5.2.1 are acceptable.	DCD Rev 18, Based on WEC letter DCP/NRC2897 dated 20100528 SCE&G Letter NND-10-0260
7582	STD,VCS	PT02	FSAR02	02.00.T / T2.0-201	COLA Part 2, FSAR Chapter 2, Section 2.0, Table 2.0-201, Sheet 5 of 5, DCD column header for HVAC Intake from the "Ground Level Containment Release Points" will be revised to omit note (8) from header notations (4), (8) to read: (4). In addition, note (8) will be removed from the list of notes at the end of Table 2.0-201 on Sheet 5 of 5.	SCE&G Letter NND-10-0260
7346	STD,VCS	PT02	FSAR02	02.00.T / T2.0-201 Sh1	COLA Part 2, FSAR Table 2.0-201, Sheet 1, Seismic, first column, to be revised from "SSE" to "CSDRS"	DCD Rev 18, Based on WEC letter DCP/NRC2668 dated 20091020

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
						SCE&G Letter NND-10-0260
7348	STD,VCS	PT02	FSAR02	02.00.T / T2.0-201 Sh1	COLA Part 2, FSAR Table 2.0-201, Sheet 1, Seismic, Fault Displacement Potential AP1000 DCD Site Parameter column entry to be revised from: "Negligible" To read: "No potential fault displacement considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures."	DCD Rev 18, Based on WEC letter DCP/NRC2897 dated 20100528 SCE&G Letter NND-10-0260
7349	STD,VCS	PT02	FSAR02	02.00.T / T2.0-201 Sh2	COLA Part 2, FSAR Table 2.0-201, Sheet 2, Soil, second parameter, to be revised from: "Maximum Allowable Dynamic Bearing Capacity for Normal Plus Safe Shutdown Earthquake (SSE)" To read: "Dynamic Bearing Capacity for Normal Plus Safe Shutdown Earthquake (SSE)"	DCD Rev 18, Based on WEC letter DCP/NRC2897 dated 20100528 SCE&G Letter NND-10-0260
7351	STD,VCS	PT02	FSAR02	02.00.T / T2.0-201 Sh3	COLA Part 2, FSAR Table 2.0-201, Sheet 3, Soil, Liquefaction Potential AP1000 DCD Site Parameter column entry to be revised from "Negligible" to read "No liquefaction considered beneath the the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures."	DCD Rev 18, Based on WEC letter DCP/NRC2897 dated 20100528 SCE&G Letter NND-10-0260
7548	VCS	PT02	FSAR02	02.02.02.02.01.01	Revise VCSNS FSAR Subsection 2.2.2.2.1.1 to add left margin annotation VCS COL 6.4-1.	SCE&G Letter NND-10-0248, Supplement to SCE&G Letter NND-10-0235 endorsement of BLN RAI LTR 169 response to RAI 06.04-008.
7304	VCS	PT02	FSAR02	02.02.03.01.03	Revise FSAR Section 2.2 to incorporate changes identifed in response to RAI 02.02.03-1 in SCE&G Letter NND-10-0012.	RAI 02.02.03-1 (Letter 075 per NND-10-0012)
7512	VCS	PT02	FSAR02	02.03 / T2.3-221	In Table 2.3-221, revise the following site limit values: 8-24 Hours: "7.45E-04" to "7.45E-05" 1-4 Days: "2.84E-04" to "2.84E-05"	SCE&G Letter NND-10-0230
7048	VCS	PT02	FSAR02	02.03.T / T2.3-227	COLA Part 2, FSAR Table 2.3-227 will be revised to include the Enhanced Fujita Scale as presented in response to RAI 19-79.	RAI 19-79 (Letter 071 Response per NND-09-0329)
7314	VCS	PT02	FSAR02	02.04.02 F / F2.4-261	Revise Subsection 2.4.2 to add new Figure 2.4-261 as shown in SCE&G Letter NND-10-0114.	RAI 02.04.13VR-2 (NND-10-0114)
7312	VCS	PT02	FSAR02	02.04.02.03	Revise the 4th paragraph of Subsection 2.4.2.3 from: As shown in Figure 2.4-210, the main plant site area is divided into four discrete subbasins, each of which has one or more distinct drainage outlets. Subbasin 1 in Figure 2.4-210 covers the western part of the site including Unit 3. Subbasin 2 overs the eastern part of the site including Unit 2. Subbasin 3 covers the northern part of the site, including the parking lot. Subbasin 4 covers the southern part of the site including the cooling tower pad.All directions mentioned in Subsection 2.4.2 .3 are with respect to the Plant North shown in Figure 2.4-210. The	RAI 02.04.13VR-2 (NND-10-0114)

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					drainage outlet for each subbasin is also shown in Figure 2.4-210. To Read: As shown in Figure 2.4-210, the main plant site area is divided into four discrete subbasins, each of which has one or more distinct drainage outlets. Subbasin 1 in Figure 2.4-210 covers the western part of the site including Unit 3. Subbasin 2 covers the eastern part of the site including Unit 2. Subbasin 3 covers the northern part of the site, including the parking lot. Subbasin 4 covers the southern part of the site including the cooling tower pad. The area located north of Subbasin 4 and east-southeast of Subbasin 2, as shown on Figure 2.4-210, was excluded from the stormwater modeling analysis as it was considered that no runoff from th is area would spill over to the adjacent subbasins due to its steeper slope and smaller drainage area. Furthermore, the areas immediately upstream of the Storm Water Basin 3 are in the supercritical flow regimes. Even with the potential addition of runoff from the excluded area, flow depths in the upstream areas adjacent to the power blocks would not be affected by backwater effects from Storm Water Basin 3. Therefore, exclusion of this area from the stormwater analysis has no effect on the assessment of maximum water levels in the power block. All directions mentioned in Subsection 2.4.2.3 are with respect to the Plant North shown in Figure 2.4-210. The drainage outlet for each subbasin is also shown in Figure 2.4-210.	
7313	VCS	PT02	FSAR02	02.04.02.03	Revise the 10th paragraph of Subsection 2.4.2.3 from: "Maximum water levels at safety-related structures of the main plant site area were calculated with the hydraulic model HEC-RAS (Version 3.1.3) developed by the U.S. Army Corps of Engineers (USACE) (Reference 236). This model uses stepwise backwater equations to estimate hydraulic flow parameters such as water levels and flow velocities for open channel systems. The steady-state option in the HEC-RAS model was used with input parameters including cross-section geometry, Manning's roughness coefficients, and flow boundary conditions." To Read: Maximum water levels at safety-related structures of the main plant site area were calculated with the hydraulic model HEC-RAS (Version 3.1.3) developed by the U.S. Army Corps of Engineers (USACE) (Reference 236). This model uses stepwise backwater equations to estimate hydraulic flow parameters such as water levels and flow velocities for open channel systems. The steady-state option in the HEC-RAS model was used with input parameters including cross-section geometry, Manning's roughness coefficients, and flow boundary conditions.The locations of the cross sections used in the HEC-RAS model are shown in Figure 2.4-261. The elevations shown in Figure 2.4-261 are referenced to NAVD88."	RAI 02.04.13VR-2 (NND-10-0114)
7315	VCS	PT02	FSAR02	02.04.02.03	Add the following paragraph to the end of Subsecton 02.04.02.03: "VCSNS Units 2 and 3 comply with state requirements for storm water basin maintenance. Additionally, station procedures will be implemented to perform walkdowns of the Units 2 and 3 yard areas prior to an anticipated heavy rain event to look for potential sources of blockage of storm drains or other inhibitors to proper storm water drainage."	RAI 02.04.13VR-2 (NND-10-0114)
7583	VCS	PT02	FSAR02	02.04.12.05	Add the following to the end of Subsection 02.04.12.05: The grading of the site and construction of Units 2 and 3 will replace the existing forest cover with buildings, parking lots, grass, gravel, etc. Overall, the post-construction land surface is less pervious and could generally result in more storm water runoff and less recharge to the aquifer. This reduced recharge could result in lower groundwater hydraulic gradients, but routing storm water runoff to the storm water basins constructed as shown in Figure 2.5.4-245 may increase recharge locally. Units 2 & 3 are designed with a storm water collection system that collects water runoff from structures and yard areas and routes it through a closed piping system, constructed of reinforced concrete and high density polyethylene, to storm water basins located onsite. Water inlets to the closed piping system are located in	Combines changes from RAI 02.04.12-6 (Letter 041 S1 per NND-10-0191) and RAI 02.04.12-6 (Letter 041 S2 per NND-10-0232)

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					parking lots, grass covered medians, yard areas, and around both power block facilities, with one or more located between Units 2 & 3 at designed low points in the final plant area grading. Buildings are constructed with a system of gutters and spouts to collect and conveywater runoff from roofs to the closed piping system. The closed piping system carries the water around the units (generally to the east or west), and then to one of the storm water basins located north or south of Units 2 & 3 (Figure 2.5.4-245). The storm water basins are located at a lower elevation than the power block areas to allow gravity flow to the basins. In order to prevent water backing up from the storm basins into the plant area, the maximum surface water elevation in the basins is lower than the final plant grade elevation (400'). The ground cover of the proposed immediate plant area will change drastically from the pre-construction conditions. These changes will reduce the amount of area available for groundwater recharge due to the addition of a large amount of impervious ground cover, a closed conduit system for handling stormwater runoff and the distance that piping system carries the water prior to discharging it into a water quality basin(s). The condition of the proposed site prior to any pre-construction operations was wooded with a groundwater level of approximately 380' (Subsection 2.4.12.5). The condition of the site following construction will include a large amount of impervious area due to the addition of structures, roads, etc. In addition, a large majority of the remaining surface area in the vicinity of the plant area will be covered in compacted gravel or similar hardscaping that will help facilitate run off away from the operating facility. As a result. the groundwater level in the vicinity of the plant area is expected to remain well below the AP1000 maximum allowable groundwater level of 398'. The ground cover following construction is expected to contain a relatively large amount of impervious cover within the immediate plant area. To quantify the approximate area of each type of groundcover expected to exist around plant, the immediate plant area is defined as the area inside of the following relative to True North: • 394' contours that run approximately North-South on either side of the proposed units as shown on Figure 2.5.4-245. • The innermost fence shown running East-West on the North side of Unit 2 and South side of Unit 3 as shown of Figure 2.5.4-245. As defined above, the immediate plant area contains approximately 42.5 acres. Included in that area is approximately 17 acres of impervious ground cover (1.5 acres of nuclear island structures founded on hard rock, 4 acres of power block structures whose foundations are located on engineered backfill, and the remaining 11.5 acres are of buildings and paved areas throughout the immediate plant area), approximately 15 acres of surface area assumed to be covered in compacted gravel or otherwise hardscaped to promote run off, and approximately 10.5 acres of grass serving as an outer most border. This combination of ground cover will shed much of the stormwater runoff to the closed Storm Drain System (DRS) for transportation away from the immediate plant area. Thus, the opportunity for infiltration recharge to increase groundwater levels in the plant area is expected to be reduced. The water collected in the DRS system is transported to unlined stormwater quality basins for sediment control. These basins will be designed as dry detention ponds per the requirements of the South Carolina Department of Health and Control (SCDHEC) "Standards for Stormwater Management and Sediment Reduction Regulation 72-307". Per these requirements, the water collected in the basins will be required to drain down within 3 days (72 hours). Therefore, the dry detention ponds used for sediment control and stormwater management should only have water in them during and immediately following a rain event. The maximum possible elevations in the basins are expected to be well below the 398' elevation as well. The basins that will be the outfalls for the DRS piping systems are anticipated to have emergency spillways that keep the maximum water elevation in each below the 395' elevation. Specifically, the top of Basin 1 (Figure 2.5.4-245) is currently designed to be 396' so that even if the outlet pipe were to be clogged, the water would overflow the basin 2' below the 398' groundwater elevation limit of the AP1000. Any rise in groundwater elevations around these basins would be temporary, away from the immediate plant area and well below the 398' elevation.	

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					Based on observation well data, the maximum expected groundwater level in the immediate plant area is 380'. VCSNS Units 2 and 3 are located on a groundwater high, with groundwater flowing radially away from the facility. Due to the small spatial extent and symmetric nature of fill placement around the power block area, the fill is not expected to significantly impact the existing groundwater pathways. FSAR Figures 2.5.4-220 through 2.5.4-223 provide cross-sections of the proposed fill material and placement around and beneath Units 2 and 3. To prevent storm water on the ground from reaching buildings, grading around buildings is designed to direct the flow of water away from the buildings. The ridge topography of the site will cause any water that may accumulate under plant structures to drain away from the plant area. With the combination of ground cover materials and the storm water collection system, any increase in groundwater level due to surface water infiltration from a heavy precipitation event is expected to be temporary, localized (i.e., around storm water basins), and remain well below the DCD maximum groundwater design elevation for hydrostatic loading of 398'.	
7700	VCS	PT02	FSAR02	02.04.12.05	Editorial correction change "of Figure 2.5.4-245" to read "on Figure 2.5.4-245" in second bullet.	Editorial correction to Change ID 7583 and combination of changes from RAI 02.04.12-6 (Letter 041 S1 per NND-10-0191) and RAI 02.04.12-6 (Letter 041 S2 per NND-10-0232) Editorial correction changes "of Figure 2.5.4-245" to read "on Figure 2.5.4-245" in second bullet.
7561	VCS	PT02	FSAR02	02.05	2. COLA Part 2, FSAR Chapter 2, Section 2.5, will be revised to add the following new Subsection 2.5.7 to read: 2.5.7 Waterproofing System VCS COL 2.5-17 This COL item is addressed in Subsection 3.8.5.1.	RAI 3.7.1-2 (Letter 53 S1 Response per SCE&G Letter NND-10-0261)
7513	VCS	PT02	FSAR02	02.05.04.12	Add the following to the end of Subsection 2.5.4.12: To establish the foundation bearing level for the Nuclear Island (NI), fill concrete will be used beneath the footprint of the NI basemat. and extending a few feet outward. The excavation around the NI and beneath other major power block structures will be backfilled with compacted granular structural fill. The relative concrete and structural fill locations are shown on FSAR Figures 2.5.4-220 through 2.5.4-223. The NI fill concrete will extend several feet (5 or 6 feet), beyond the footprint of the NI. Concrete fill will be used between the bottom of the NI foundation and the finish grade on sound rock. Based on the top of Layer V (Sound Rock) contours (FSAR Figure 2.5.4-202), the top of sound rock occurs generally at EI. 360 +/- 5 ft beneath the NI at Units 2 and 3, but it is approximately 17 ft lower (EL 343 ft) at the northeast corner of the Unit 2 NI and approximately 12 ft higher (EL 372 ft) beneath the southern part of the Unit 2 NI. The NI areas will be excavated in sound rock to approximately EL 357 ft, where required, to allow a minimum 3 ft thickness of fill concrete and mud mat beneath the NI basemats. The fill concrete will be approximately 17 ft thick beneath the northeastern corner of the Unit 2 basemat.	RAI 02.05.04-36 (Letter 056 S1 per NND-10-0231)

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					American Concrete Institute (ACI) defines mass concrete as "any volume of concrete with dimensions large enough to require that measures be taken to cope with generation of heat from hydration of the cement and attendant volume change to minimize cracking." The definition is intentionally vague because many factors, including the concrete mix design, the dimensions, the type of the placement, and the curing methods, affect whether or not cracking will occur. ACI 207, "Mass Concrete," prepared by ACI Committee 207, governs the design and construction of mass concrete. Typically, there are two common concerns associated with thermal cracks in mass concrete. They are: (1) the maximum temperature inside a concrete pour and (2) the maximum temperature difference between the hottest spot and the surface of a concrete pour. Specifications of mass concrete typically limit the maximum temperature to 155°F and the maximum temperature difference between the interior and the surface to 36°F, so that early-age thermal cracks in mass concrete will be minimized. It is a common practice to limit the least dimension of each concrete pour so that the temperature and temperature difference of the pour can stay within their respective limits. Since the northeastern corner under the Unit 2 basemat is expected to require approximately 17 feet of fill concrete, according to the definition of mass concrete in ACI 207, "Mass Concrete", the fill concrete under the NI of Unit 2 is a mass concrete. A thermal control plan considering the geometry of Unit 2 fill concrete, the proposed 5,000 psi strength, total volume of fill concrete placement, and rate of concrete production, will be prepared to help ensure that the rule-of-thumb temperature limits will not be exceeded. The thermal control plan, based on the ACI 207 guidelines for preventing thermal cracking in concrete, will have the following elements: • Use well-graded aggregate and Type I and/or II cement in the concrete mix. • Because of its relatively high strength specification, the fill concrete will likely have a high content of Portland cement substitutes, such as Class F flyash and/or slag, to minimize the heat of hydration. • In anticipation of variations in elevation in sound rock surface, the minimum thickness of fill concrete will be set at 3 feet, which includes the 6-inch layer of mud mat. • Even with the heat of hydration in the design mix minimized, it may still require the concrete to be placed in relatively thin lifts to avoid cracking. Thus, the maximum thickness of each concrete lift will be set at about five feet. • Concrete will be placed using a step technique to minimize the live face of concrete, thus minimizing the chance for cold joints. • Exposed surfaces of each concrete lift will be insulated, if required. • When another lift is required on top of an existing lift, the top lift will be poured only after the bottom lift has enough time to properly cool down. • Concrete placing temperature will be controlled as necessary by use of ice, chilled water, shading aggregate piles, spraying coarse aggregate for evaporative cooling, and scheduling placements (such as at night) to take	

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					advantage of coolest temperatures. • Planned vertical joints in each concrete lift will be properly treated. • Planned horizontal joints between two concrete lifts will be properly treated.	
7668	VCS	PT02	FSAR02	02.05.04.14	2. COLA Part 2, FSAR Chapter 2, Section 2.5, will be revised to add the following Subsection 2.5.4.14 to read and renumber 2.5.4.14 References Subsection to 2.5.4.15: 2.5.4.14 Waterproofing System VCS COL 2.5-17 This COL item is addressed in Subsection 3.8.5.1.	Editorial Correction to Change ID 7561 and RAI 3.7.1-2 (Letter 53 S1 Response per SCE&G Letter NND-10-0261) to put the COL information item in the correct subsection.
6634	VCS	PT02	FSAR02	02.05.05	Revise lead in statement of FSAR 2.5.5 from: Add the following text to the end of Subsection 2.5.5. to read: Add the following text to the end of DCD Subsection 2.5.5.	Editorial for consistency
6635	VCS	PT02	FSAR02	02.05.06	Revise lead in statement of FSAR 2.5.6 from: Add the following text to the end of Subsection 2.5.6. to read: Add the following text to the end of DCD Subsection 2.5.6.	Editorial for consistency
7686	VCS	PT02	FSAR02	Table 2.2-209 / Sheet 2	Revise Footnote (g), item 4, to read "(4) the control room air exchange rate--that is, the time it would take the outdoor air to replace the indoor air in the control room after the vapor cloud has passed the control room. Based upon this evaluation, the 8-hour TWA would not be exceeded."	Editorial correction to RAI 02.02.03-1 (Letter 075 per NND-10-0012). Note change in RAI response of "past" to "passed"
PT02 - FSAR03						**26 COLA Changes**
7354	STD,VCS	PT02	FSAR03	03.06.04.01	COLA Part 2, FSAR Chapter 3, Subsection 3.6.4.1, will be revised from: Replace the last paragraph in DCD Subsection 3.6.4.1 with the following text. A pipe rupture hazard analysis is part of the piping design. It is used to identify postulated break locations and layout changes, support design, whip restraint design, and jet shield design. The final design for these activities will be completed prior to fabrication and installation of the piping and connected components. The as-built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load. To read: Replace the last paragraph in DCD Subsection 3.6.4.1 with the following text.	DCD Revision 18, COL-SER-OI-Ch03 S6 response to OI 03.06-001 (item 2 SNC Letter ND-10-0801)

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					The as-designed pipe rupture hazards evaluation is made available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets protected by associated mitigation features (Reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate. A pipe rupture hazards analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The report will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design. The pipe whip restraint and jet shield design includes the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design will be completed prior to installation of the piping and connected components. The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load (in accordance with DCD Tier 1 Table 3.3-6, item 8). This COL item is also addressed in Subsection 14.3.3.	
7356	STD,VCS	PT02	FSAR03	03.07.04.04	COLA Part 2, Chapter 3, Section 3.7.4.4 will be revised to add the following text to the end of the existing FSAR Subsection 3.7.4.4. In addition, the procedures address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool walls, and provide for appropriate corrective actions to be taken if needed (such as repositioning the racks or analysis of the as-found condition).	DCD Rev 18, Based on WEC letters DCP/NRC2609 dated 20090831
7563	VCS	PT02	FSAR03	03.08.05.01	COLA Part 2, FSAR Chapter 3, Subsection 3.8.5.1, will be revised to add a new LMA of VCS COL 2.5-17 with the following existing statement to read: VCS COL 2.5-17 A sheet type waterproofing material will be used for both the horizontal and vertical surfaces under seismic Category I structures. The material will be qualified by test, with commercial grade dedication and lab testing to achieve a minimum coefficient of friction (COF) of 0.70.	DCD Rev 18, RAI 3.7.1-2 (Letter 53 S1 Response per SCE&G Letter NND-10-0261)
7644	STD,VCS	PT02	FSAR03	03.09.03.01.02	2. COLA Part 2, FSAR Chapter 3, Subsection 3.9.3.1.2, will be revised under the heading of General, from: The pressurizer surge line is monitored at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters. This monitoring occurs during the hot functional testing and first fuel cycle. The resulting monitoring data is evaluated to verify that the pressurizer surge line is within the bounds of the analytical temperature distributions and displacements. The pressurizer surge line monitoring activities include the following methodology and requirements: To read: The pressurizer surge line is monitored at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters. This monitoring occurs during the hot functional testing and first fuel cycle. The resulting monitoring data is evaluated to verify that the pressurizer surge line is within the bounds of the analytical temperature distributions and displacements.	VEGP-RAI-LTR 057 response to RAI 03.12-002 item 2, SNC Ltr ND-10-1263

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					Subsequent AP1000 plants (after the first AP1000 plant) confirm that the heatup and cooldown procedures are consistent with the pertinent attributes of the first AP1000 plant surge line monitoring. In addition, changes to the heatup and cooldown procedures consider the potential impact on stress and fatigue analyses consistent with the concerns of NRC Bulletin 88-11. The pressurizer surge line monitoring activities include the following methodology and requirements:	
7357	STD,VCS	PT02	FSAR03	03.09.06.02.02	Revise the last sentence of the third paragraph following the paragraph in FSAR Subsection 3.9.6.2.2 containing subheading "Other Power-Operated Valve Operability Tests" from: ... The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air-and other power-operated valves included in the IST program. For example: To read: ... The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air-and other power-operated valves included in the IST program. For example, key lessons learned addressed in the AOV program include:	COL-SER-OI-Ch03 response to OI 03.09-04 (SNC Ltr ND-09-2015)
7358	STD,VCS	PT02	FSAR03	03.09.06.02.02	Revise the last sentence of the third bulleted paragraph following the paragraph in FSAR Section 3.9.6.2.2 containing subheading "Other Power-Operated Valve Operability Tests" from: • Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If the margin between component capability and design-basis requirements has not been previously determined, dynamic testing will be performed to establish a baseline and to determine these margins. To read: • Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required, based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.	COL-SER-OI-Ch03 response to OI 03.09-04 (SNC Ltr ND-09-2015)
7359	STD,VCS	PT02	FSAR03	03.09.06.02.02	Subsection 3.9.6.2.2 containing subheading "Other Power-Operated Valve Operability Tests" from: • Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement. have the potential to affect high risk valve functional performance. To read: • Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.	COL-SER-OI-Ch03 response to OI 03.09-04 (SNC Ltr ND-09-2015)
7360	STD,VCS	PT02	FSAR03	03.09.06.02.02	Add the paragraph below as the last paragraph of FSAR Subsection 3.9.6.2.2 prior to the subheading "Check Valve Tests": The attributes of the AOV testing program described above, to the extent that they apply to and can be	COL-SER-OI-Ch03 response to OI 03.09-04 (SNC Ltr ND-09-2015)

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					implemented on other safety-related power-operated valves, such as electro-hydraulic valves, are applied to those other power-operated valves.	
7364	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Subsection 3.9.6.2.2 will be revised as follows: Add the following paragraph following the paragraph with the heading "Active MOV Test Frequency Determination": Maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) are established so as not to exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.	COL-SER-OI-Ch03 S3 response to OI 03.09-003 (f) item 2 (SNC Ltr ND-10-0393) and S7 response to OI 03.09-03(f) item 2 (SNC Ltr ND-10-0949)
7365	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Subsection 3.9.6.2.2 will be revised as follows: Insert the following paragraph as the last paragraph under the sub-heading of "Power-Operated Valve Operability Tests" (following the previously added paragraph) and just before the sub-heading "Check Valve Tests" in DCD Subsection 3.9.6.2.2: Successful completion of the preservice and IST of MOVs, in addition to MOV testing as required by 10 CFR 50.55a, demonstrates that the following criteria are met for each valve tested: (i) valve fully opens and/or closes as required by its safety function; (ii) adequate margin exists and includes consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load-sensitive MOV behavior, and margin for degradation; and (iii) maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) does not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.	COL-SER-OI-Ch03 S3 response to OI 03.09-003 (f) item 3 (SNC Ltr ND-10-0393) and S7 response to OI 03.09-03(f) item 3 (SNC Ltr ND-10-0949)
7366	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Subsection 3.9.6.2.2 will be revised as follows: (1) Revise the COLA insert entitled "Active MOV Test Frequency Determination" from: Active MOV Test Frequency Determination -The ability of a valve to meet its design basis functional requirements (i.e. required capability) is verified during valve qualification testing as required by procurement specifications. Requirements for qualification testing of power-operated active valves are included in procurement specifications. Valve qualification testing measures valve actuator output capability. Actuator output capability is compared to the valve's required capability defined in procurement specifications, establishing functional margin; that is, that increment by which the MOV's actual output capability exceeds the capability required to operate the MOV under design basis conditions. DCD Subsection 5.4.8 discusses valve functional design and qualification requirements. The inservice test frequency is determined as required by the ASME OM Code, Code Case OMN-1. Valve functional margin is evaluated to account for anticipated time-related changes in performance, accounting for applicable uncertainties in the analysis. If the evaluation shows that the functional margin will be reduced to less than established acceptance criteria within the established test interval, the test interval is decreased to less than the time for the functional margin to decrease below acceptance criteria. If there is not sufficient data to determine test frequency as described above, the test frequency is limited to not exceed two (2) refueling cycles or three (3) years, whichever is longer, until sufficient data exist to extend the test frequency. Maximum test frequency shall not exceed 10 years, and appropriate justification is provided for any increased test interval. This is to ensure that each MOV in the IST program will have adequate margin (including consideration for aging-related degradation) to remain operable until the next scheduled test, regardless of its risk categorization or safety significance. Uncertainties associated with performance of these periodic verification tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed. To read: Active MOV Test Frequency Determination -The ability of a valve to meet its design basis functional	COL-SER-OI-Ch03 S7 response to OI 03.09-03(f) SNC Ltr ND-10-0949

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					requirements (i.e. required capability) is verified during valve qualification testing as required by procurement specifications. Valve qualification testing measures valve actuator actual output capability. The actuator output capability is compared to the valve's required capability defined in procurement specifications, establishing functional margin; that is, that increment by which the MOV's actual output capability exceeds the capability required to operate the MOV under design basis conditions. DCD Subsection 5.4.8 discusses valve functional design and qualification requirements. The initial inservice test frequency is determined as required by ASME OM Code Case OMN-1, Revision 1 (Reference 202). The design basis capability testing of MOVs utilizes guidance from Generic Letter 96-05 and the JOG MOV Periodic Verification PV Program. Valve functional margin is evaluated following subsequent periodic testing to address potential time-related performance degradation, accounting for applicable uncertainties in the analysis. If the evaluation shows that the functional margin will be reduced to less than established acceptance criteria within the established test interval, the test interval is decreased to less than the time for the functional margin to decrease below acceptance criteria. If there is not sufficient data to determine test frequency as described above, the test frequency is limited to not exceed two (2) refueling cycles or three (3) years, whichever is longer, until sufficient data exist to extend the test frequency. Appropriate justification is provided for any increased test interval, and the maximum test interval shall not exceed 10 years. This is to ensure that each MOV in the IST program will have adequate margin (including consideration for aging-related degradation, degraded voltage, control switch repeatability, and load-sensitive MOV behavior) to remain operable until the next scheduled test, regardless of its risk categorization or safety significance. Uncertainties associated with performance of these periodic verification tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.	
7367	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised as follows: Revise the third roadmap (for clarity) from: Add the following as a new last paragraph under the heading "Manual/Power-Operated Valve Tests": During valve exercise tests, the necessary valve obturator movement is determined while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflects change of obturator position. To read: Add the following paragraph after the fifth paragraph under the heading "Manual/Power-Operated Valve Tests": During valve exercise tests, the necessary valve obturator movement is verified while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflects change of obturator position.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(c) item 1 SNC Ltr ND-10-0949
7368	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised as follows: Revise the fourth insert in Subsection 3.9.6.2.2, and the roadmap, from: Add the following at the end of the last sentence of the paragraph containing the subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2: , and for motor-operated valves the JOG MOV PV study (Reference 201) and ASME Code Case OMN-1 Revision 1 (Reference 202). Table 13.4-201 provides milestones for the MOV program implementation.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(c) item 1 SNC Ltr ND-10-0949

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					To read: Add the following sentence as the last sentence of the paragraph containing the subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2: Table 13.4-201 provides the milestones for the MOV program implementation.	
7369	STD,VCS	PT02	FSAR03	03.09.06.02.02	(1) COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised as follows: (iii) Delete the fifth COLA insert and roadmap in Subsection 3.9.6.2.2 that reads: Revise the first sentence of the second paragraph under the paragraph with subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2 to read as follows: Static and dynamic testing with diagnostic measurements will be performed on these valves as described below.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(c) item 1 SNC Ltr ND-10-0949
7370	STD,VCS	PT02	FSAR03	03.09.06.02.02	(1) COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised as follows: (iv) Revise the sixth COLA insert in Subsection 3.9.6.2.2, from: Insert the following as the last sentence in the paragraph under the bulleted item titled "Risk Ranking" in OCD Subsection 3.9.6.2.2: Guidance for this process is outlined in the JOG MOV PV Study, MPR-2524-A (Reference 201). To read: Insert the following as the last sentence in the paragraph under the bulleted item titled "Risk Ranking" in DCD Subsection 3.9.6.2.2: Guidance for this process is outlined in the JOG MOV PV Study, MPR-2524-A.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(c) item 1 SNC Ltr ND-10-0949
7371	STD,VCS	PT02	FSAR03	03.09.06.02.02	(1) COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised as follows: (v) Revise COLA Subsection 3.9.6.2.2 paragraph beginning with the subheading "Other Power-Operated Valve Operability Tests," from: Other Power-Operated Valve Operability Tests -Power-Operated valves other than active MOVs are exercised quarterly in accordance with ASME OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies. Active and passive power-operated valves upon which operability testing may be performed are identified in DCD Table 3.9-16. To read: Other Power-Operated Valve Operability Tests -Power-Operated valves other than active MOVs are exercised quarterly in accordance with ASME OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(c) item 1 SNC Ltr ND-10-0949
7372	STD,VCS	PT02	FSAR03	03.09.06.02.02	DCD Chapter 3, Subsection 3.9.6.2.2, will be revised in COLA Subsection 3.9.6.2.2 as follows to address COL Information Item 3.9-4: NOTE: The following changes are in consideration of and in addition to the changes made by Westinghouse in their responses to DCD SER Open Items on Section 3.9.6. Insert new second sentence of the paragraph containing the subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2 (immediately following the first sentence of the DCD paragraph) to read:	COL-SER-OI-Ch03 S7 response to OI 03.09-002(c) item 2 SNC Ltr ND-10-0949

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					Power-Operated Valve Operability Tests - The safety-related, power-operated valves (POVs) are required by the procurement specifications to have the capabilities to perform diagnostic testing to verify the capability of the valves to perform their design basis safety functions. The POVs include the motor-operated valves. (ii) Add the left margin annotation "STD COL 3.9-4" for the above change.	
7373	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised to include the following new paragraph with a left margin annotation (LMA) of STD COL 3.9-4: Add the following new paragraph under the heading "Other Valve Inservice Tests" following the Explosively Actuated Valves paragraph in DCD Subsection 3.9.6.2.2: Industry and regulatory guidance is considered in development of the IST program for squib valves. In addition, the IST program for squib valves incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions.	VEGP RAI LTR 056 response to RAI 03.09.06-001
7600	STD,VCS	PT02	FSAR03	03.09.06.02.02	COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.2.2, will be revised to change "incorporate" to "incorporates" in the new paragraph added via Qb 7217 from: In addition, the IST program for squib valves incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions. To read: In addition, the IST program for squib valves incorporates lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions.	Editorial revision to VEGP RAI LTR 056 response to RAI 03.09.06-001
7375	STD,VCS	PT02	FSAR03	03.09.06.03	(1) Revise the inserted paragraph of COLA Part 2, FSAR Chapter 3, Subsection 3.9.6.3, Relief Requests, from: The IST Program described herein utilizes Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants" (Reference 202). Code Case OMN-1 establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves, in lieu of the requirements set forth in ASME OM Code Subsection ISTC. Implementation of the program described in Code Case OMN-1 will require request for relief, unless Code Case OMN-1, Revision 1, is approved by the NRC in Regulatory Guide 1.192, or the case has been incorporated into the ASME OM Code on which the IST program is based, and that Code is approved in 10 CFR 50.55a(b). To read: The IST Program described herein utilizes Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants" (Reference 202). Code Case OMN-1 establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves, in lieu of the requirements set forth in ASME OM Code Subsection ISTC.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(b) item 1 SNC Ltr ND-10-0949
7376	STD,VCS	PT02	FSAR03	03.09.06.03	(2) Add new discussion at end of current FSAR Subsection 3.9.6.3 (with the same LMA as the current FSAR Subsection 3.9.6.3) as follows: OMN-1, Alternative Rules for the Preservice and Inservice Testing of Certain MOVs Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in Light Water Reactor Power Plants," establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves in lieu of the requirements set forth in OM Code Subsection ISTC. However, Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," June 2003, has not yet endorsed OMN-	COL-SER-OI-Ch03 S7 response to OI 03.09-002(b) item 2 SNC Ltr ND-10-0949

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					1, Revision 1. Code Case OMN-1, Revision 0, has been determined by the NRC to provide an acceptable level of quality and safety when implemented in conjunction with the conditions imposed in Regulatory Guide 1.192. NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," recommends the implementation of OMN-I by all licensees. Revision 1 to OMN-1 represents an improvement over Revision 0, as published in the ASME OM-2004 Code. OMN-1 Revision 1 incorporates the guidance on risk-informed testing of MOVs from OMN-11, "Risk-Informed Testing of Motor-Operated Valves," and provides additional guidance on design basis verification testing and functional margin, which eliminates the need for the figures on functional margin and test intervals in Code Case OMN-1. The IST Program implements Code Case OMN-1, Revision 1, in lieu of the stroke-time provisions specified in ISTC-5120 for MOVs, consistent with the guidelines provided in NUREG-1482, Revision 1, Section 4.2.5. Regulatory Guide 1.192 states that licensees may use Code Case OMN-1, Revision 0, in lieu of the provisions for stroke-time testing in Subsection ISTC of the 1995 Edition up to and including the 2000 Addenda of the ASME OM Code when applied in conjunction with the provisions for leakage rate testing in ISTC-3600 (1998 Edition with the 1999 and 2000 Addenda). Licensees who choose to apply OMN-1 are required to apply all of its provisions. The IST program incorporates the following provisions from Regulatory Guide 1.192: (1) The adequacy of the diagnostic test interval for each motor-operated valve (MOV) is evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of OMN-1. (2) The potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement. (3) Risk insights are applied using MOV risk ranking methodologies accepted by the NRC on a plant-specific or industry-wide basis, consistent with the conditions in the applicable safety evaluations. (4) Consistent with the provisions specified for Code Case OMN-11 the potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement. Compliance with the above items is addressed in Section 3.9.6.2.2. Code Case OMN-1, Revision 1, is considered acceptable for use with OM Code-2001 Edition with 2003 Addenda. Finally, consistent with Regulatory Guide 1.192, the benefits of performing any particular test are balanced against the potential adverse effects placed on the valves or systems caused by this testing.	
7378	STD,VCS	PT02	FSAR03	03.09.08.02	3. COLA Part 2, FSAR Chapter 3, Subsection 3.9.8.2, will be revised from: Add the following text after the second paragraph in DCD Subsection 3.9.8.2. Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in DCD Subsection 3.9.3.1.2) is completed after the construction of the piping systems and prior to fuel load. To read: Add the following text after the second paragraph in DCD Subsection 3.9.8.2. Design specifications and design reports for ASME Section III piping are made available for NRC review. Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in DCD Subsection 3.9.3.1.2) is completed by the COL holder after the construction of the piping systems and prior to fuel load (in accordance with DCD Tier 1 Section 2 ITAAC line items for the applicable systems).	COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 3 SNC Letter ND-10-0801

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7645	STD,VCS	PT02	FSAR03	03.09.08.05	3. COLA Part 2, FSAR Chapter 3, Subsection 3.9.8.5, will be revised from: This COL item is addressed in Subsection 3.9.3.1.2. To read: This COL item is addressed in Subsection 3.9.3.1.2 and Subsection 14.2.9.2.22.	VEGP-RAI-LTR 057 response to RAI 03.12-002 item 3, SNC Ltr ND-10-1263
7380	STD,VCS	PT02	FSAR03	03.09.08.07	4. COLA Part 2, FSAR Chapter 3, Subsection 3.9.8.7, will be added to read: --- 3.8.9.7 As-Designed Piping Analysis --- Add the following text at the end of DCD Subsection 3.9.8.7. The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in DCD Table 3.9-19 is made available for NRC review. This COL item is also addressed in Subsection 14.3.3. ---	COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 4 SNC Letter ND-10-0801
7382	STD,VCS	PT02	FSAR03	03.09.09	(3) Revise Subsection 3.9.9, REFERENCES, as follows: From: 201. Joint Owners Group (JOG) Motor Operated Valve Periodic Verification Program Summary, MPR 2524-A, ADAMS ML063490199, November 2006. To read: 201. Not used.	COL-SER-OI-Ch03 S7 response to OI 03.09-002(b) item 3 SNC Ltr ND-10-0949
7747	STD,VCS	PT02	FSAR03	03.09.T / T3.9-201	COLA, Part 2, FSAR Chapter 3, Table 3.9-201, second column, 8th entry down will be revised from: SGS APP-SGS-PH-11Y0065 005B To read: SGS APP-SGS-PH-11Y0065 L005B	Editorial correction for consitent identification of line numbers associated with snubbers.
PT02 - FSAR05						**4 COLA Changes**
7383	STD,VCS	PT02	FSAR05	05.03.02.06	Revise COLA Part 2, FSAR Section 5.3.2.6 inserted information, 8th inserted paragraph that begins with "Standard Charpy impact..." from "...establish and..." to "...establish an..."	Editorial
7565	VCS	PT02	FSAR05	05.04.07.01.02.03	Revise VCSNS FSAR Subsection 5.4.7 as follows: --- Replace the second bulleted item in DCD Subsection 5.4.7.1.2.3 with the following: VCS DEP 2.0-2 • The component cooling water system supply temperature to the normal residual heat removal system heat exchangers is based on an ambient design wet bulb temperature of no greater than 87.3°F (100 year return estimate of 2-hour duration). The 87.3°F value is assumed for normal	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)

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					conditions and transients that start at normal conditions. The steaming prevention function is evaluated assuming the ambient wet bulb temperature is at the maximum safety value for the site. During plant operation, maximum IRWST temperature is reduced below 120 °F whenever necessary by circulating IRWST water through one of the RNS heat exchangers, and removing the heat through the CCS and SWS. Since the RNS heat exchangers are not being used to remove decay heat with the plant at power, at least one is available for IRWST heat removal. Only one train of CCS (pump and heat exchanger) and one train of SWS (pump, strainer, and cooling tower cell) are normally in operation with the plant at power. There is sufficient margin in CCS pump flow capacity and motor size, and in CCS heat exchanger UA, to valve in one of the RNS heat exchangers and remove IRWST heat by directing CCS flow through the heat exchanger and transferring the excess heat to the SWS cooling tower. CCS temperature rises slightly above the normal full power CCS temperature during this evolution, but does not approach the maximum allowable value of 100 °F. Prevention of IRWST steaming following high pressure heat removal operations with the Passive Residual Heat Removal (PRHR) heat exchanger is accomplished in the same manner, by lining up both RNS heat exchangers to the CCS and the IRWST. CCS is delivered to the RNS heat exchangers at a temperature consistent with the maximum safety ambient wet bulb temperature and the CCS and SWS heat duty and flow rates. Cooling is assumed to begin two hours after reactor trip, with decay heat appropriate for that time after the event. Calculations performed to determine the maximum IRWST temperature achieved following a high pressure heat removal event using the PRHR heat exchanger assumed CCS temperature is determined by use of a maximum safety ambient wet bulb temperature value of 87.4 °F. The maximum predicted IRWST liquid temperature is 201°F. Therefore, it can be concluded that IRWST cooling performance(prevention of steaming) is acceptable. ----------	
7664	VCS	PT02	FSAR05	05.04.07.01.02.03	Revise VCSNS FSAR Subsection 5.4.7 as follows: ---------- Replace the second bulleted item in DCD Subsection 5.4.7.1.2.3 with the following: VCS DEP 2.0-2 • The component cooling water system supply temperature to the normal residual heat removal system heat exchangers is based on an ambient design wet bulb temperature of no greater than 87.3°F (100 year return estimate of 2-hour duration). The 87.3°F value is assumed for normal conditions and transients that start at normal conditions. The steaming prevention function is evaluated assuming the ambient wet bulb temperature is at the maximum safety	Editorial Correction of Change ID 7565 to add reference citation to the end of inserted text.

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					value for the site. During plant operation, maximum IRWST temperature is reduced below 120 °F whenever necessary by circulating IRWST water through one of the RNS heat exchangers, and removing the heat through the CCS and SWS. Since the RNS heat exchangers are not being used to remove decay heat with the plant at power, at least one is available for IRWST heat removal. Only one train of CCS (pump and heat exchanger) and one train of SWS (pump, strainer, and cooling tower cell) are normally in operation with the plant at power. There is sufficient margin in CCS pump flow capacity and motor size, and in CCS heat exchanger UA, to valve in one of the RNS heat exchangers and remove IRWST heat by directing CCS flow through the heat exchanger and transferring the excess heat to the SWS cooling tower. CCS temperature rises slightly above the normal full power CCS temperature during this evolution, but does not approach the maximum allowable value of 100 °F. Prevention of IRWST steaming following high pressure heat removal operations with the Passive Residual Heat Removal (PRHR) heat exchanger is accomplished in the same manner, by lining up both RNS heat exchangers to the CCS and the IRWST. CCS is delivered to the RNS heat exchangers at a temperature consistent with the maximum safety ambient wet bulb temperature and the CCS and SWS heat duty and flow rates. Cooling is assumed to begin two hours after reactor trip, with decay heat appropriate for that time after the event. Calculations performed to determine the maximum IRWST temperature achieved following a high pressure heat removal event using the PRHR heat exchanger assumed CCS temperature is determined by use of a maximum safety ambient wet bulb temperature value of 87.4 °F. The maximum predicted IRWST liquid temperature is 201°F. Therefore, it can be concluded that IRWST cooling performance(prevention of steaming) is acceptable (Reference 202).	
7572	VCS	PT02	FSAR05	05.04.16	Revise VCSNS FSAR Subsection 5.4.16 to add the following new reference: 202. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010.	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
PT02 - FSAR06						**11 COLA Changes**
7386	STD,VCS	PT02	FSAR06	06.01.02.01.06	Revise COLA Part 2, FSAR Chapter 6, Section 6.1.2.1.6, to incorporate to read: Replace the third paragraph under the subsection titled "Service Level I and Service Level III Coatings" within DCD Subsection 6.1.2.1.6 with the following information. During the design and construction phase the coatings program associated with selection, procurement and application of safety related coatings is performed to applicable quality standards. The requirements for the coating program are contained in certified drawings and/or standards and specifications controlling the coating processes of the designer (Westinghouse) (these design documents will be available prior to the procurement and application of the coating material by the constructor of the plant). Regulatory Guide 1.54 and ASTM	VEGP-VOL-CH06 response to STD COL 06.01-002 SNC Ltr ND-10-1264

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					D5144 (Reference 201) form the basis for the coating program. During the operations phase, the coatings program is administratively controlled in accordance with the quality assurance program implemented to satisfy 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements. The coatings program provides direction for the procurement, application, and monitoring of safety related coating systems. Prior to initial fuel loading, a consolidated plant coating program will be in place to address procurement, application, inspection, and monitoring (maintenance) of those coating system(s) for the life of the plant. Coating system monitoring requirements for the containment coating systems are based on ASTM D5163 (Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant," and ASTM D7167 (Reference 203), "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality assurance requirements. Replace the second sentence of the third paragraph under the subsection titled "Service Level II Coatings" within DCD Subsection 6.1.2.1.6 with the following information. Coating system inspection and monitoring requirements for the Service Level II coatings used inside containment will be performed in accordance with a program based on ASTM D5144 (Reference 201), "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants," and the guidance of ASTM D5163 (Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality requirements.	
7566	VCS	PT02	FSAR06	06.02.01.01.03	Revise VCSNS FSAR Section 6.2 to add the following information with LMA VCS DEP 2.0-2: --- 6.2.1.1.3 Design Evaluation --- Add the following information after the fourth paragraph of DCD Subsection 6.2.1.1.3. The maximum safety non-coincident wet bulb temperature for VCSNS Units 2 and 3 is increased from 86.1 °F to 87.3 °F, however there are no impacts on the performance of the safety systems. ---	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7666	VCS	PT02	FSAR06	06.02.02.03	Revise VCSNS FSAR Section 6.2 to add the following information with LMA VCS DEP 2.0-2: --- 6.2.2.3 Safety Evaluation --- Add the following information at the end of DCD Subsection 6.2.2.3. There are no changes to the AP1000 design required to address any safety issues associated with the VCSNS Units 2 and 3 increased maximum safety wet bulb temperature of 87.3 °F. The peak containment pressure at the maximum safety wet bulb temperature of 87.3 °F for the VCSNS Units 2 and 3 site is bounded by the results of the current AP1000 analysis. The pressure decay curve for the containment utilizing the VCSNS Units 2 and 3 safety wet bulb value of 87.3 °F is the same as the containment response for wet bulb temperatures equal to the standard maximum safety wet bulb value(Reference 201). ---	Correction to RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254) (Corrects standard "maximum safety wet bulb" to "VCSNS Units 2 and 3 safety wet bulb" in Section 6.2.2.3, second paragraph, Also Adds "Reference 201" to end of text).
7573	VCS	PT02	FSAR06	06.02.07	Revise VCSNS FSAR Section 6.2 as follows:	RAIs 6.2.1-1 and

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					--- 6.2.7 REFERENCES --- 201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010. ---	9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7684	VCS	PT02	FSAR06	06.04	Revise VCSNS FSAR Section 6.4 to add the following information with LMA VCS DEP 2.0-2: --- Add the following information after the second paragraph of DCD Subsection 6.4. Based on system design margin of the VBS, the MCR temperature and humidity at the higher VCSNS maximum safety wet bulb temperature will remain at or below the desired design points during normal operation (Reference 201). ---	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254), Includes correction to add Reference 201 to end of added text.
7683	VCS	PT02	FSAR06	06.04.01.01	Revise VCSNS FSAR Section 6.4 to add the following information with LMA VCS DEP 2.0-2: --- 6.4.1.1 Main Control Room Design Basis --- Add the following information after the last paragraph of DCD Subsection 6.4.1.1: The VBS system maintains design conditions in the MCR during all normal and accident conditions when the VBS system is operational. The LCCWS also serves the RNS and CVS pump room coolers. The nominal refrigeration capacity of each of the air-cooled chillers used in the LCCWS is 322 tons at an ambient dry bulb temperature of 115 °F (Reference 201). ---	RAIs 6.2.1-1 S2 and 9.2.2-1 S2 (Letters 079 S2 and 081 S2 Response per NND-10-0269), Includes Correction to Add Reference 201 to end of inserted text
7389	STD,VCS	PT02	FSAR06	06.04.03	3. Revise COLA Part 2, Chapter 6, Subsection 6.4.3, to include the following new final paragraphs under the STD COL LMA for the existing paragraph: The procedures and training address the toxic chemical events addressed in Sections 2.2 and 6.4 consistent with the guidance provided in regulatory position C.5 of Regulatory Guide 1.78, including arrangements with Federal, State, and local agencies or other cognizant organizations for the prompt notification of the nuclear power plant when accidents involving hazardous chemicals occur within five miles of the plant. The procedures include the conduct of periodic surveys of stationary and mobile sources of hazardous chemicals affecting the evaluations consistent with the guidance provided in regulatory position 2.5 of Regulatory Guide 1.196. The procedures include appropriate reviews of the configuration of the control room envelope and habitability systems consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196. The procedures also include periodic assessments of the control room habitability systems' material condition, configuration controls, safety analyses, and operating and maintenance procedures consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196. Procedures for testing and maintenance are consistent with the design requirements of the DCD including the guidance provided in regulatory position 2.7.1 of Regulatory Guide 1.196.	RAI LTR 168 response to RAI 06.04-007 item 3
7550	VCS	PT02	FSAR06	06.04.04	Revise VCSNS FSAR Subsection 6.4.4 to add left margin annotation STD COL 6.4-1.	SCE&G Letter NND-10-0248, Supplement to SCE&G Letter NND-10-0235 endorsement of BLN RAI LTR 169

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						response to RAI 06.04-008.
7551	VCS	PT02	FSAR06	06.04.07	Revise VCSNS FSAR Subsection 6.4.7 as follows: --- 6.4.7 COMBINED LICENSE INFORMATION --- VCS COL 6.4-1 This COL Item is addressed in Subsections 2.2.2.2.1.1 and 6.4.4.2. --- STD COL 6.4-2 This COL Item is addressed in Subsection 6.4.3. --- STD COL 6.4-1 This COL Item is addressed in Subsection 6.4.4. ---	SCE&G Letter NND-10-0248, Supplement to SCE&G Letter NND-10-0235 endorsement of BLN RAI LTR 169 response to RAI 06.04-008.
7574	VCS	PT02	FSAR06	06.04.08	Revise VCSNS FSAR Section 6.4 as follows: --- 6.4.8 REFERENCES --- 201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010. ---	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7540	VCS	PT02	FSAR06	6.4-201.T / T6.4-201	Revise FSAR Table 6.4-201 as shown in SCE&G Letter NND-10-0235 dated June 24, 2010.	SCE&G Letter NND-10-0235 endorsement of BLN RAI LTR 169 response to RAI 06.04-008
PT02 - FSAR07						**7 COLA Changes**
7394	STD,VCS	PT02	FSAR07	07.01	2. COLA Part 2, FSAR Chapter 7, Section 7.1, will be revised from: 7.1 INTRODUCTION This section of the referenced DCD is incorporated by reference with no departures or supplements. To read (with an LMA of STD COL 7.1-1 for the new subsection 7.1.6.1): 7.1 INTRODUCTION This section of the referenced DCD is incorporated by reference with the following departures and/or supplements. --- 7.1.6.1 Setpoint Calculations for Protective Functions The Setpoint Program described in Technical Specifications Section 5.5 provides the appropriate controls for update of the instrumentation setpoints following completion of the calculation of setpoints for protective functions and the reconciliation of the setpoints against the final design.	DCD Rev 18, VEGP-VOL-CH07 response to 07.01-001 item 2 SNC Ltr ND-10-1118, VEGP-VOL-CH07 S1 response to 07.01-001 item 2 SNC Ltr ND-10-1266
7641	STD,VCS	PT02	FSAR07	07.05	3. COLA Part 2, FSAR Chapter 7, Section 7.5, will be revised for Subsections 7.5.2 and 7.5.3.5 to read (with LMAs for both subsections of STD COL 7.5-1):	DCD Rev 18, VEGP-VOL-CH07 S1

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					7.5.2 VARIABLE CLASSIFICATONS AND REQUIREMENTS Add the following paragraph at the end of DCD Subsection 7.5.2. FSAR Table 7.5-201 supplements DCD Table 7.5-1 and provides variable data shown in the DCD Table as "site specific." 7.5.3.5 Type E Variables Add the following paragraph at the end of DCD Subsection 7.5.3.5. FSAR Table 7.5-202 supplements DCD Table 7.5-8 and provides variable data shown in the DCD Table as "site specific."	item 3, SNC Ltr ND-10-1266
7642	STD,VCS	PT02	FSAR07	07.05	4. COLA Part 2, FSAR Chapter 7, Section 7.5, will be revised to add Subsection 7.5.5 with LMAs of both STD COL 7.5-1 and VCS COL 7.5-1: 7.5.5 COMBINED LICENSE INFORMATION This COL item is addressed in Subsection 7.5.2 and Table 7.5-201, and in Subsection 7.5.3.5 and Table 7.5-202.	DCD Rev 18, VEGP-VOL-CH07 S1 item 4, SNC Ltr ND-10-1266
7547	VCS	PT02	FSAR07	07.05 / 7.5-1	Revise FSAR Section 7.5 to read as follows: "This section of the referenced DCD is incorporated by reference with the following departures and/or supplements. Add the following paragraph at the end of Subsection 7.5.2. FSAR Table 7.5-201 supplements DCD Table 7.5-1 and provides variable data shown in DCD Table 7.5-1 as "site specific." Add the following paragraph at the end of Subsection 7.5.3.5. FSAR Table 7.5-202 supplements DCD Table 7.5-8 and provides variable data shown in DCD Table 7.5-1 as "site specific."	VEGP RAI response to LTR 006
7396	STD,VCS	PT02	FSAR07	07.05.05	3. COLA Part 2, FSAR Chapter 7, Section 7.5, will be revised to add Subsection 7.5.5: 7.5.5 COMBINED LICENSE INFORMATION STD COL 7.5-1 This COL item is addressed in Subsections 7.5.2 and 7.5.3.5.	DCD Rev. 18, VEGP-VOL-CH07 response to 07.04-001 item 3 SNC Ltr ND-10-1118
7709	STD,VCS	PT02	FSAR07	07.05.05	Add horizontal separator line after FSAR Section 7.5.5.	Editorial
7398	STD,VCS	PT02	FSAR07	07.05.T	5. COLA Part 2, FSAR Chapter 7, Section 7.5, LMAs for Tables 7.5-201 and 7.5-202 will be revised from "VCS SUP 7.5-1" to "VCS COL 7.5-1."	DCD Rev 18, VEGP-VOL-CH07 response to 07.04-001 item 5 SNC Ltr ND-10-1118 VEGP-VOL-CH07 S1 item 5, SNC Ltr ND-10-1266
PT02 - FSAR08						**3 COLA Changes**

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7306	VCS	PT02	FSAR08	08.02.02	Revise the fifth paragraph of FSAR Subsection 8.2.2 from: The grid stability analysis confirmed that the interface requirements for steady-state load, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and limiting under frequency value for the RCP have been satisfied. To Read: Table 8.2-201 shows that the interface requirements for steady-state load, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and limiting under frequency value for the RCP have been satisfied.	RAI 01-4 (Letter 055 S2 per NND-10-0100)
7307	VCS	PT02	FSAR08	08.02.02	Revise FSAR Subsection 8.2.2 to remove the last paragraph: "Grid stability analysis includes the interface items identified in DCD Table 1.8-1, Item 8.2."	RAI 01-4 (Letter 055 S2 per NND-10-0100)
7308	VCS	PT02	FSAR08	08.02.T / T8.2-201	Revise FSAR Subsection 8.2 to include new Table 8.2-201 "Grid Stability Interface Evaluation" with LMA of VCS COL 8.2-2.	RAI 01-4 (Letter 055 S2 per NND-10-0100)
PT02 - FSAR09						**14 COLA Changes**
7569	VCS	PT02	FSAR09	09.01.03.01.03.01	Revise VCSNS FSAR Subsection 9.1 to add the following information with LMA VCS DEP 2.0-2: --- 9.1.3.1.3.1 Partial Core --- Add the following information at the end of the third bullet in DCD Subsection 9.1.3.1.3.1. SFS performance following restart after a normal refueling is affected by a change in maximum safety wet bulb temperature. Calculations confirm that spent fuel pool temperature remains below 115 °F with a CCS supply temperature of 97 °F at the specified pool spent fuel loading condition and decay time on the fuel fraction just replaced during the previous 17 day refueling outage. While the maximum CCS temperature expected for VCSNS Units 2 and 3 is 97.3 °F, an increase of 0.3 °F in CCS supply temperature will produce a similar increase in the spent fuel pool maximum temperature; therefore, the requirement to maintain spent fuel temperature below 120 °F is met with margin. ---	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7689	VCS	PT02	FSAR09	09.01.03.01.03.01	Revise VCSNS FSAR Subsection 9.1 to add the following information with LMA VCS DEP 2.0-2: --- 9.1.3.1.3.1 Partial Core --- Add the following information at the end of the third bullet in DCD Subsection 9.1.3.1.3.1. SFS performance following restart after a normal refueling is affected by a change in maximum safety wet bulb temperature. Calculations confirm that spent fuel pool temperature remains below 115 °F with a CCS supply temperature of 97 °F at the specified pool spent fuel loading condition and decay time on the fuel fraction just replaced during the previous 17 day refueling outage. While the maximum CCS temperature expected for VCSNS Units 2 and 3 is 97.3 °F, an increase of 0.3 °F in CCS supply temperature will produce a similar increase in the spent fuel pool maximum temperature; therefore, the requirement to maintain spent fuel temperature below 120 °F is met with margin (Reference 201). ---	Correction to Change ID 7569 and RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254) to add Reference 201 to end of text addition.
7399	STD,VCS	PT02	FSAR09	09.01.04.04	COLA Part 2, FSAR Chapter 9, Subsection 9.1.4.4, will be revised to add the following sentence at the end of the	COL-SER-OI-Ch09

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					STD COL 9.1-5 description. The light load handling program, including system inspections, is implemented prior to receipt of fuel onsite.	response to OI 09.01-002 SNC Letter ND-09-2078
7400	STD,VCS	PT02	FSAR09	09.01.05.04	COLA Part 2, FSAR Chapter 9, Subsection 9.1.5.4, will be revised to add the following sentence at the end of the STD COL 9.1-5 description. The overhead heavy load handling program, including system inspections, is implemented prior to receipt of fuel onsite.	COL-SER-OI-Ch09 response to OI 09.01-003 (SNC Letter ND-09-2078)
7402	STD,VCS	PT02	FSAR09	09.01.06	1. COLA Part 2, FSAR Chapter 9, Subsection 9.1.6, will be revised to add the following to the end of the STD COL 9.1-7 description. The program will include the methodology and acceptance criteria for the tests listed and provide corrective action requirements based on vendor recommendations and industry operating experience. The program will be implemented through plant procedures. Metamic Monitoring Acceptance Criteria: • Verification of continued presence of the boron is performed by neutron attenuation measurement. A decrease of no more than 5% in Boron-10 content, as determined by neutron attenuation, is acceptable. This is equivalent to a requirement for no loss in boron within the accuracy of the measurement. • Coupons are monitored for unacceptable swelling by measuring coupon thickness. An increase in coupon thickness at any point of no more than 10% of the initial thickness at that point is acceptable. Changes in excess of either of the above two acceptance criteria are investigated under the corrective action program and may require early retrieval and measurement of one or more of the remaining coupons to provide validation that the indicated changes are real. If the deviation is determined to be real, an engineering evaluation is performed to identify further testing or any corrective action that may be necessary. Additional parameters are examined for early indications of the potential onset of Metamic degradation that would suggest a need for further attention and possibly a change in the coupon withdrawal schedule. These include visual inspection for surface pitting, blistering, cracking, corrosion or edge deterioration, or unaccountable weight loss in excess of the measurement accuracy.	COL-SER-OI-Ch09 S1 response to OI 09.01-001 item 1 SNC Letter ND-10-0781
7575	VCS	PT02	FSAR09	09.01.07	Revise VCSNS FSAR Section 9.1 as follows: --- 9.1.7 REFERENCES --- 201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010. ---	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7570	VCS	PT02	FSAR09	09.02.02.01.02.01	Revise VCSNS FSAR Subsection 9.2.2.1 as follows: --- Replace the first bullet item in the criteria for normal operation in DCD Subsection 9.2.2.1.2.1 with the following information. VCS DEP 2.0-2 • The component cooling water supply temperature to plant components is not more than 100°F assuming a 100-year return estimate of 2-hour duration wet bulb temperature of 87.3°F for service water cooling (per Table 2.0-201).	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					The most limiting component cooled by the CCS, the RCP motor cooling system, has been designed to operate for at least 6 hours continually with cooling water supplied at temperatures up to 100 °F. The performance of the standard AP1000 CCS and SWS for single cooling water train, full power operation at a maximum safety wet bulb temperature of 87.4 °F has demonstrated the highest CCS temperature achieved at these conditions is 97.4 °F, for a period of less than 2 hours. As ambient wet bulb temperature decreases, the CCS temperature follows and will return to below 95 °F with ambient wet bulb temperature slightly lower than 84 °F, assuming nominal performance of both the CCS and SWS. Since the definition of the maximum normal wet bulb temperature value is the seasonal 1 % exceedance value observed at the site, the annual total operating time for which CCS temperatures could exceed 95 °F is less than 30 hours per year, for periods of a few hours at most. The maximum CCS temperature of 97.3 °F is bounded by the maximum allowable cooling water temperature for Reactor Coolant Pumps (the most limiting component) and the increase in maximum safety wet bulb temperature is therefore acceptable on this basis. ---	
7690	VCS	PT02	FSAR09	09.02.02.01.02.01	Revise VCSNS FSAR Subsection 9.2.2.1 as follows: --- Replace the first bullet item in the criteria for normal operation in DCD Subsection 9.2.2.1.2.1 with the following information. VCS DEP 2.0-2 • The component cooling water supply temperature to plant components is not more than 100°F assuming a 100-year return estimate of 2-hour duration wet bulb temperature of 87.3°F for service water cooling (per Table 2.0-201). The most limiting component cooled by the CCS, the RCP motor cooling system, has been designed to operate for at least 6 hours continually with cooling water supplied at temperatures up to 100 °F. The performance of the standard AP1000 CCS and SWS for single cooling water train, full power operation at a maximum safety wet bulb temperature of 87.4 °F has demonstrated the highest CCS temperature achieved at these conditions is 97.4 °F, for a period of less than 2 hours. As ambient wet bulb temperature decreases, the CCS temperature follows and will return to below 95 °F with ambient wet bulb temperature slightly lower than 84 °F, assuming nominal performance of both the CCS and SWS. Since the definition of the maximum normal wet bulb temperature value is the seasonal 1 % exceedance value observed at the site, the annual total operating time for which CCS temperatures could exceed 95 °F is less than 30 hours per year, for	Correction to Change ID 7570 and RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254) to add Reference 201 to end of text

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					periods of a few hours at most. The maximum CCS temperature of 97.3 °F is bounded by the maximum allowable cooling water temperature for Reactor Coolant Pumps (the most limiting component) and the increase in maximum safety wet bulb temperature is therefore acceptable on this basis (Reference 201). ----------	
7571	VCS	PT02	FSAR09	09.02.07.02.04	Revise VCSNS FSAR to add Subsection 9.2.7.2.4 to add the following information with LMA VCS DEP 2.0-2: ---------- 9.2.7.2.4 System Operation ---------- Add the following information at the end of the first paragraph under "Normal Operation" in DCD Subsection 9.2.7.2.4. The increased heat load produced by operation at the higher VCSNS maximum safety ambient wet bulb temperature of 87.3 °F can be accommodated within the available capacity margin of the chiller units, without impacting the LCCWS or supporting systems design or plant operation. Cooling coil design calculations indicate that during operation at the standard plant design temperatures (115 °F dry bulb, 86.1 °F wet bulb), the VBS air handling unit has cooling coil and system margin. ----------	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7691	VCS	PT02	FSAR09	09.02.07.02.04	Revise VCSNS FSAR to add Subsection 9.2.7.2.4 to add the following information with LMA VCS DEP 2.0-2: ---------- 9.2.7.2.4 System Operation ---------- Add the following information at the end of the first paragraph under "Normal Operation" in DCD Subsection 9.2.7.2.4. The increased heat load produced by operation at the higher VCSNS maximum safety ambient wet bulb temperature of 87.3 °F can be accommodated within the available capacity margin of the chiller units, without impacting the LCCWS or supporting systems design or plant operation. Cooling coil design calculations indicate that during operation at the standard plant design temperatures (115 °F dry bulb, 86.1 °F wet bulb), the VBS air handling unit has cooling coil and system margin (Reference 201). ----------	Correction to RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254) to add Reference 201 to end of text.
7317	VCS	PT02	FSAR09	09.02.09.02.02	Replace Subsection 9.2.9.2.2 in its entirety as follows: ---------- 9.2.9.2.2 Component Description ---------- Add the following text under the Waste Water Retention Basin paragraph of DCD Subsection 9.2.9.2.2 and add Basin Transfer Pumps as follows: The waste water retention basin is constructed using formed concrete and is a lined basin constructed such that its contents, dissolved or suspended, do not penetrate the liner and leach into the ground. Each Unit's Waste Water Retention Basin (WWRB) is located in the yard area outside of each Unit's respective Turbine Building. The WWRB is designed to allow entrained solids to settle and allow for chemical treatment of effluent concentrations required for release prior to discharge to the blowdown sump. The configuration and size of the waste water retention basin allows settling of solids larger than 10 microns that may be suspended in the waste water stream. Waste water can be sampled prior to discharge from the waste water retention basin.	RAI 09.03.03-1 (Letter 080 per NND-10-0147)

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					Each WWRB is divided into two separate compartments, which allows one compartment to be out of service while the other compartment is available. Each compartment discharges to a pump sump. A level transmitter located in each WWRB pump sump provides an alarm signal in the Main Control Room when the sump level(s) reach predetermined set points. Basin Transfer Pumps In the event of oily waste leakage into the retention basin, a recirculation line is provided to recycle the oil/water waste from the basin to the oil separator. The WWRB transfer pumps are located in pump sumps adjacent to each compartment. The pumps are manually started and interlocked to stop based on sump level. There are two (one per sump) 100% capacity transfer pumps for each WWRB. The transfer pumps are sized to meet the maximum expected influent flow. The normal pump discharge flowpath is to the blowdown sump. Flow can also be directed to the other Unit's WWRB. Blowdown Sump/Plant Outfall The blowdown sump is a concrete structure and is open to the atmosphere. It is a common sump and accepts waste water from both Units' WWRBs, CWS cooling tower blowdown from both Units and sanitary waste effluent. In the absence of CWS cooling tower blowdown, RWS supplies an alternate source of dilution water. The outfall pipe is sized with adequate capacity to gravity drain the blowdown sump at the highest anticipated influent flow rate. Wastewater and blowdown effluent from the blowdown sump drains by gravity to Parr Reservoir via the plant outfall piping. Location of the plant outfall routing is shown on FSAR Figure 1.1-202.	
7576	VCS	PT02	FSAR09	09.02.13	Revise VCSNS FSAR Section 9.2.13 to add the following reference: 201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010.	RAIs 6.2.1-1 and 9.2.2-1 (Letters 079 and 081 Response per NND-10-0254)
7309	VCS	PT02	FSAR09	09.05.02.05.01	Revise FSAR Subsection 9.5.2.5.1 from: "This COL Item is addressed in Part 2, Section F "Emergency Communications" of the Emergency Plan." To Read: "The Emergency Notification System (ENS) and the Emergency Response Data System (ERDS) are both powered normally by the 120V-ac power system. In the event of a loss of the ac power system, the systems are automatically switched over to the diesel backed, non-Class 1E dc and uninterruptable power supply systems. Additional information regarding emergency communication systems can be found in Part 2, Section F "Emergency Communications" of the Emergency Plan."	RAI 09.05.02-7 (Letter 073 per NND-10-0104)
7310	VCS	PT02	FSAR09	09.05.02.05.02	Revise FSAR Subsection 9.5.2.5.2 from: "This COL Item is addressed in Part 2, Section F "Emergency Communications" of the Emergency Plan." To Read: "The primary system used for communication with state and county officials during an emergency is the Electric Switch System Exchange (ESSX). VCSNS employs additional backup communication systems to the ESSX system including the use of the Private Branch Exchange (PBX) telephone system, local commercial telephone system, satellite telephones, and an 800 MHz radio system. In the event of the failure of one of the primary systems, the communicator manually initiates communications using one of the backup systems as described in	RAI 09.05.02-8 (Letter 073 per NND-10-0104)

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					the Emergency Implementing Procedures. The Implementing Procedures provide the details for the communications transfer should the primary equipment fail or otherwise be determined to be unacceptable. The 800 MHz system serves as the crisis management radio system between VCSNS onsite teams and state and county officials. Details of the primary and secondary communication systems are provided in Section F of the VCSNS Emergency Plan."	
PT02 - FSAR11						**4 COLA Changes**
7585	VCS	PT02	FSAR11	11.02.01.02.04	Revise VCSNS FSAR Subsection 11.2.1.2.4 as shown in SCE&G Letter NND-10-0265, dated July 8, 2010.	SCE&G Letter NND-10-0265
7586	VCS	PT02	FSAR11	11.02.06	Revise VCSNS FSAR Subsection 11.2.6 as shown in SCE&G Letter NND-10-0265, dated July 8, 2010.	SCE&G Letter NND-10-0265
7316	VCS	PT02	FSAR11	11.02.T / T 11.2-204	Revise Table 11.2-204 to include corrected values shown in SCE&G Letter NND-10-0128.	Voluntary Update of Table 11.2-204 (NND-10-0128)
7403	STD,VCS	PT02	FSAR11	11.05.07	Revise COLA Part 2, FSAR Chapter 11, Subsections 11.5.7 and 11.5.8 to be renumbered from: 11.5.7 Combined License Information, and 11.5.8 References To read: 11.5.8 Combined License Information, and 11.5.9 References. Also, in the lead-in statement to introduces the COLA FSAR References section, revise the statement from: "Add the following subsection after DCD Subsection 11.5.7" to read "Add the following subsection after DCD Subsection 11.5.8"	DCD Rev 18, Based on WEC letter DCP/NRC2492 dated 20090522
PT02 - FSAR12						**4 COLA Changes**
7318	VCS	PT02	FSAR12	12.04.01.09.02	Revise the 2nd paragraph of Subsection 12.4.1.9.2 from: "Direct radiation from the Unit 1 containment and other plant buildings is negligible. Routine operational thermo-luminescent dosimeter (TLD) measurements at the site boundary for Unit 1 show that the annual doses are comparable to the preoperational annual dose rates. For conservatism, the annual direct dose to a construction worker at Unit 2 or 3 is assumed to be 1 mrem per year. Small quantities of monitored airborne effluents are normally released from the Unit 1 from the waste gas decay tank, reactor building purges, and oil incineration. The construction workers are assumed to be exposed to the gaseous and liquid doses from routine operation of Unit 1." To Read: Direct radiation from the Unit 1 containment and other plant buildings is negligible. Routine operational thermo-luminescent dosimeter (TLD) measurements at the site boundary for Unit 1 show that the annual doses are comparable to the preoperational annual dose rates. For conservatism, the direct dose rate from Unit 1 in the construction area for Units 2 and 3 is assumed to be 1 mrem per year. Small quantities of monitored airborne effluents are normally released from the Unit 1 from the waste gas decay tank, reactor building purges, and oil incineration. The construction workers are assumed to be exposed to the gaseous and liquid doses from routine operation of Unit 1.	RAI 12.03-12.04-3 (Letter 072 per NND-10-0151)
7537	VCS	PT02	FSAR12	12.04.01.09.02	Revise the 4th paragraph of Subsection 12.4.1.9.2 from: "For Unit 2, the radiation exposure at the site boundary is considered in DCD Section 12.4.2. As stated in that section, direct radiation from the containment and other plant buildings is negligible. Additionally, there is no contribution from refueling water since the refueling water is stored inside the containment instead of in an outside storage tank. For conservatism, the annual dose to a construction worker at Unit 3 is assumed to be 1 mrem per year."	RAI 12.03-12.04-3 (Letter 072 per NND-10-0151)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					To Read: "For Unit 2, the radiation exposure at the site boundary is considered in DCD Section 12.4.2. As stated in that section, direct radiation from the containment and other plant buildings is negligible. Additionally, there is no contribution from refueling water since the refueling water is stored inside the containment instead of in an outside storage tank. For conservatism, the dose rate from Unit 2 in the construction area for Unit 3 is assumed to be 1 mrem per year."	
7319	VCS	PT02	FSAR12	12.04.01.09:03	Revise the 3rd paragraph of Subsection 12.4.1.9.3 from: "The methodology used to calculate the doses to construction workers from normal effluent releases complies with the guidance provided in Regulatory Guide 1.109. Construction worker doses were estimated by use of LADTAP II computer code (NUREG/CR-4013). The total effective dose equivalent (TEDE), which is the sum of the deep dose equivalent (DDE) and the committed effective dose equivalent (CEDE), was determined from the LAPTAP II results. The annual TEDE dose was corrected for the actual time the construction workers are onsite by multiplying by the ratio of hours worked per year to hours in a year." To Read: "The methodology used to calculate the doses to construction workers from normal effluent releases complies with the guidance provided in Regulatory Guide 1.109. Construction worker doses were estimated by use of LADTAP II computer code (NUREG/CR-4013). The total effective dose equivalent (TEDE), which is the sum of the deep dose equivalent (DDE) and the committed effective dose equivalent (CEDE), was determined based on the LADTAP II results. The annual TEDE dose was corrected for the actual time the construction workers are onsite by multiplying by the ratio of hours worked per year to hours in a year."	RAI 12.03-12.04-3 (Letter 072 S1 per NND-10-0151)
7320	VCS	PT02	FSAR12	12.04.01.09.03	Add the following paragaraph to the end of Subsection 12.4.1.9.3: "When adjusted for an occupancy time of 2000 hours per year, the direct, gaseous, and liquid doses from Unit 1 are 0.23, 0.27, and 0.0020 mrem TEDE, respectively, yielding a total annual dose of 0.50 mrem TEDE. The occupancy-adjusted direct, gaseous, and liquid doses from Unit 2 are 0.23, 0.48, and 0.067 mrem TEDE, respectively, resulting in a total of 0.77 mrem TEDE. Therefore, the total annual dose to the Unit 3 construction worker from Units 1 and 2 is 1.3 mrem TEDE."	RAI 12.03-12.04-3 (Letter 072 S1 per NND-10-0151)
PT02 - FSAR13						**25 COLA Changes**
7404	STD,VCS	PT02	FSAR13	13.03	2. COLA Part 2, FSAR Chapter 13, Section 13.3, first sentence will be revised to read: STD COL 13.3-1 The emergency planning information is submitted to the Nuclear Regulatory Commission as a separate licensing document and is incorporated by reference. (see Table 1.6-201).	VEGP-VOL-Ch13 response to STD 13.03-01 item 2 SNC Ltr ND-10-1036
7650	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 11	4. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 11, Non-Licensed Plant Staff Training Program, will be revised from: (portions applicable to SNM) To read: (portions applicable to radioactive material)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 4 SNC Ltr ND-10-1305
7651	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 11	5. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 11, Non-Licensed Plant Staff Training Program, will be revised to add the following reference under the "Program Source" column.	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 5

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					10 CFR 70.22	SNC Ltr ND-10-1305
7652	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 11	6. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 11, Non-Licensed Plant Staff Training Program, will be revised to add the following reference under the "Implementation Requirement" column. 10 CFR 70.22(a)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 6 SNC Ltr ND-10-1305
7653	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 14	7. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 14, Emergency Planning, Program Title column, will be revised from: (portions applicable to SNM) To read: (portions applicable to radioactive material)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 7 SNC Ltr ND-10-1305
7654	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 14	8. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 14, Emergency Planning, Program Source column, will be revised from: 10 CFR 30.32 10 CFR 40.31 To read: 10 CFR 30.32(i)(3) 10 CFR 40.31(j)(3) 10 CFR 70.22(i)(3)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 8 SNC Ltr ND-10-1305
7655	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 14	9. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 14, Emergency Planning, Implementation - Requirement column, will be revised from: 10 CFR 30.32(a) 10 CFR 40.31(a) To read: 10 CFR 30.32(i)(1) 10 CFR 40.31(j)(1) 10 CFR 70.22(i)(1)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 9 SNC Ltr ND-10-1305
7656	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 15	10. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 15, Physical Security Program, Program Title column, will be revised from: (portions applicable to SNM) To read: (portions applicable to radioactive material)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 10 SNC Ltr ND-10-1305
7657	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 15	11. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 15, Physical Security Program, will be revised to add the following reference under the "Program Source" column. 10 CFR 73.1	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 11 SNC Ltr ND-10-1305
7658	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 15	12. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 15, Physical Security Program, will be revised to add the following reference under the "Implementation Requirement" column. 10 CFR 73.1(a)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 12 SNC Ltr ND-10-1305

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7407	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 20	1. COLA Part 2, FSAR, Table 13.4-201, item 20 will be revised. (Refer to the final submitted letter for the complete revision)	VEGP-RAI-LTR 049 S1 response to RAI 13.06-035 item 1 SNC Ltr ND-10-1230 RAI 13.6.1-1 (Letter 076 S1 Response per NND-10-240)
7699	VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 20	COLA Part 2, FSAR, Table 13.4-201 item 20, will be revised under the "Requirement' column for the Program "FFD Program for Operation" to read: " 10 CFR Part 26, Subparts A - I,N, and O, except for individuals listed in § 26.4(b), who are not subject to §§ 26.205 – 209" Note: change of "listen" to "listed"	Grammatical Correction to RAI 13.6.1-1 (Letter 076 S1 Response per NND-10-240)
7405	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 21	COLA Part 2, FSAR Section 13.4, Table 13.4-201, item 21, Cyber Security Milestone column to be revised from "Prior to initial fuel load" to read "Prior to receipt of fuel onsite (protected area)"	VEGP RAI LTR 51 Erratum 1 - SNC ND-10-0773
7647	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 8	1. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 8, Fire Protection Program, Program Title column, will be revised from: (portions applicable to SNM) To read: (portions applicable to radioactive material)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 1 SNC Ltr ND-10-1305
7648	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 8	2. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 8, Fire Protection Program, will be revised to add the following reference under the "Program Source" column. 10 CFR 70.22 (portions applicable to radioactive material)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 2 SNC Ltr ND-10-1305
7649	STD,VCS	PT02	FSAR13	13.04.T / T13.4-201 Item 8	3. COLA Part 2, FSAR Chapter 13, Section 13.4, Table 13.4-201, item 8, Fire Protection Program, will be revised to add the following reference under the "Implementation Requirement" column. 10 CFR 70.22(a)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 3 SNC Ltr ND-10-1305
7408	STD,VCS	PT02	FSAR13	13.05.01	COLA Part 2, FSAR, Subsection 13.5.1, Administrative Procedures, 7th paragraph, will be revised by adding new bulleted text at the end of the current set of bullets, to read: • A process for implementing the safety/security interface requirements of 10 CFR 73.58.	VEGP RAI LTR 55 response to RAI 13.06-036 RAI 13.06.01-17 (Letter 083 Response per SCE&G Letter NND-10-0180)
6211	STD,VCS	PT02	FSAR13	13.06	1. Change COLA Part 2, FSAR, Section 13.6, Security, third and fourth paragraphs: Current text: An operational program following the guidance of NRC endorsed NEI 04-04 Revision 1 (Reference 202) will be implemented prior to fuel load.	VOL-SEC-CYBER-20090811 item 1 - SUPERSEDES a portion of Qb 4992

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					Table 13.4-201 provides milestones for security program implementation. To read: The Cyber Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document to fulfill the requirements contained in 10 CFR 52.79(a)(36) and 10 CFR 73.54. The Cyber Security Plan will be maintained in accordance with the requirements of 10 CFR 52.98. The Plan is withheld from public disclosure pursuant to 10 CFR 2.390. Table 13.4-201 provides milestones for security program and cyber security program implementation.	
7409	STD,VCS	PT02	FSAR13	13.06	2. COLA Part 2, FSAR Chapter 13, Section 13.6, first paragraph, will be revised from: The Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements of 10 CFR 52.79(a)(35) and 52.79(a)(36). To read: The Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements of 10 CFR 52.79(a)(35) and 52.79(a)(36) and is incorporated by reference (see Table 1.6-201).	VEGP-VOL-Ch13 response to STD COL 13.06-01 item 2 SNC Ltr ND-10-1036
7410	STD,VCS	PT02	FSAR13	13.06	3. COLA Part 2, FSAR Section 13.6, second paragraph, to be revised to include an IBR statement for the Cyber Security Plan (CSP) from: The Cyber Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document to fulfill the requirements contained in 10 CFR 52.79(a)(36) and 10 CFR 73.54. To read: The Cyber Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document to fulfill the requirements contained in 10 CFR 52.79(a)(36) and 10 CFR 73.54 and is incorporated by reference (see Table 1.6-201).	VEGP-VOL-Ch13 response to STD COL 13.06-01 item 3 SNC Ltr ND-10-1036 Note this modifies Change ID 7409
7552	STD,VCS	PT02	FSAR13	13.06.01	1. COLA Part 2, FSAR, Section 13.6.1, Combined License Information Item, will be revised from: Information for the Security Plan portion of this COL item is addressed in Section 13.6. To Read: [Reviewer's Note: The current left-margin annotation (LMA), STD COL 13.6-1, applies to both sentences.] Information for the Security Plan portion of this COL item is addressed in Section 13.6. Information for the Physical Security ITAAC portion of this COL item is addressed in Section 14.3.2.3.2.	DCD Rev 18, Based on WEC letter DCP/NRC2719 dated 20091216 VEGP RAI LTR 047 S2 response to RAI 14.03.12-001 item 1 SNC Ltr ND-10-0886 RAI 13.6.1-13 (Letter 077 Response per SCE&G Letter NND-10-0236)
6213	STD,VCS	PT02	FSAR13	13.06.02	COLA Part 2, FSAR. Chapter 13, Subsection 13.6.2, will be revised to remove Reference 202.	Changes to address Security Regulation revisions (complements Qb 4992)
7660	VCS	PT02	FSAR13	13.07	1. COLA Part 2, FSAR, Section 13.7, will be revised from: The Fitness for Duty (FFD) Program is implemented and maintained in two phases; the construction phase program and the operating phase program. The construction and operations phase programs are implemented	VEGP RAI LTR 049 response to RAI 13.06-033 item 1 RAI 13.06.01-1

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					as identified in Table 13.4-201 . The construction phase program is consistent with NEI 06-06 (Reference 201). The workforce population subject to random testing during construction is determined on a weekly basis by averaging the total number of active construction badges over each preceding seven day period. The random selection from each week's workforce population is identified by a standard computer generated random number generator using this number of active badges as the range of numbers considered in the weekly random testing selection. The operations phase program is consistent with 10 CFR Part 26. To read: [Reviewer's Note: The first, second, and fourth paragraphs are annotated with left-margin annotation (LMA) STD-SUP-13.7-1; the third paragraph, including bullets, is annotated with LMA VCS-SUP-13.7-1] The Fitness for Duty Program (FFD) is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site. In general, two different FFD programs will be implemented: a construction FFD program and an operations FFD program. The construction and operations phase programs are illustrated in Table 13.4-201. The construction FFD program is consistent with NEI 06-06 (Reference 201). NEI 06-06 applies to persons constructing or directing the construction of safety- and security-related structures, systems, or components performed onsite where the new reactor will be installed and operated. Management and oversight personnel, as further described in NEI 06-06, and security personnel prior to the receipt of special nuclear material in the form of fuel assemblies (with certain exceptions) will be subject to the operations FFD program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and O. At the establishment of a protected area, all persons who are granted unescorted access will meet the requirements of an operations FFD program. Prior to issuance of a Combined License, the construction FFD program at a new reactor construction site for those subject to Subpart K will be reviewed and revised as necessary should substantial revisions occur to either NEI 06-06 following NRC endorsement or the requirements of 10 CFR Part 26. --- The following site-specific information is provided: • The FFD program for the construction site, as defined in NEI 06-06, will be administered under a VCSNS-approved Shaw Stone & Webster (Shaw) program. The 10 CFR Part 26 requirements will be implemented for the construction site area based on the descriptions provided in Table 13.4-201. • Construction Workers & First Line Supervisors (Shaw employees and subcontractors) will be covered by a VCSNS-approved Shaw FFD Program (elements Subpart K). • SCE&G employees and SCE&G subcontractor's construction management and oversight personnel will be covered by the VCSNS Unit 1 Operations FFD Program (elements Subpart A - H, N and O) and Shaw's employees and Shaw's subcontractors construction management and oversight personnel will be covered by a VCSNS-approved Shaw FFD Program (elements Subpart A - H, N and O). • VCSNS security personnel will be covered by the VCSNS Unit 1 FFD Operations Program (elements Subpart A - H, N and O) and Shaw's security personnel will be covered by the VCSNS-approved Shaw FFD Program (elements Subpart A - H, N and O). This coverage is applicable from the start of construction activities to the earlier of (1) the receipt of SNM in the form of fuel assemblies, (2) the establishment of a protected area, or (3) the 10 CFR 52.103(g) finding.	(Letter 076 Response per NND-10-0082)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					• VCSNS FFD Program personnel will be covered by the VCSNS Unit 1 Operations FFD Program and Shaw's FFD Program personnel will be covered by the VCSNS-approved Shaw FFD Program (elements Subpart A, B, D-H, N, O, and C per Licensee discretion). • VCSNS security personnel protecting fuel assemblies will be covered by the VCSNS Unit 1 Operations FFD Program (elements Subpart A - I, N and O). • Personnel required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF) by Emergency Plans and procedures when that requirement is in effect will be covered by the VCSNS Unit 1 Operations FFD Program, except for subsections 26.205-209. --- The operations phase FFD program is consistent with the applicable subparts of 10 CFR Part 26.	
7413	STD,VCS	PT02	FSAR13	13.07.01	2. COLA Part 2, FSAR, Section 13.7.1, will be revised from: 201. Nuclear Energy Institute "Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites," NEI 06-06, Revision 4, February 2009. To read: [Reviewer's Note: This reference is standard supplemental information and is included with the STD-SUP-13.7-1 LMA from the fourth paragraph of Section 13.7.] 201. Nuclear Energy Institute, "Fitness for Duty Program Guidance for New Reactor Power Plant Construction Sites," NEI 06-06, Revision 5, August 2009 (ML092430016).	VEGP RAI LTR 049 response to RAI 13.06-033 item 2 (SNC Letter ND-10-0461) RAI 13.06.01-1 (Letter 076 Response per NND-10-0082)
7414	STD,VCS	PT02	FSAR13	13.07.01	2. COLA Part 2, FSAR, Section 13.7.1, Reference 201, will be revised from: 201. Nuclear Energy Institute, "Fitness for Duty Program Guidance for New Reactor Power Plant Construction Sites," NEI 06-06, Revision 5, August 2009 (ML092430016). To read: 201. Nuclear Energy Institute, "Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites," NEI 06-06, Revision 5, August 2009 (ML092430016).	Editorial correction of document title provided in VEGP RAI LTR 049 response to RAI 13.06-033 item 2
PT02 - FSAR14						**16 COLA Changes**
7415	STD,VCS	PT02	FSAR14	14.02.01	Revise COLA Part 2, FSAR Subsection 14.2.1 inserted information, first inserted paragraph that begins with "FSAR Section 14.2 provides..." from "...(ITP}..." to "...(ITP)..."	Editorial
7416	STD,VCS	PT02	FSAR14	14.02.02	Revise COLA Part 2, FSAR Subsection 14.2.2 from: "The PT&O organization structure (organizational chart) is included in Startup Administrative Manual." To read: "The PT&O organization structure (organizational chart) is included in the Startup Administrative Manual."	Editorial
7417	STD,VCS	PT02	FSAR14	14.02.02.02	Revise COLA Part 2, FSAR Subsection 14.2.2.2 inserted information, second inserted paragraph that begins with "The training program..." to omit an unintended space from "...program /procedures..." to "...program/procedures..."	Editorial
7418	STD,VCS	PT02	FSAR14	14.02.02.04	Revise COLA Part 2, FSAR Subsection 14.2.2.4 inserted information, bullet for Construction Group, first sentence from "...Balance-of-Plant engineering..." to "...Balance of Plant (BOP) engineering..."	Editorial
7419	STD,VCS	PT02	FSAR14	14.02.02.04	Revise COLA Part 2, FSAR Subsection 14.2.2.4 inserted information, bullet for Construction Services Training	Editorial

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					Group, first sentence from "...Accordance..." to "...accordance..."	
7420	STD,VCS	PT02	FSAR14	14.02.02.04	Revise COLA Part 2 FSAR Subsection 14.2.2.4 inserted information, bullet for Construction Services Quality Group, change from 10 CFR 50.55e ... to... 10 CFR 50.55(e)	Editorial
7421	STD,VCS	PT02	FSAR14	14.02.03.01.04	Revise COLA Part 2, FSAR Subsection 14.2.3.1.4, from "...52.1 03g..." to "...52.103(g)..." and from "...52.103g..." to "...52.103(g)..."	Editorial
7422	STD,VCS	PT02	FSAR14	14.02.03.01.05	Revise COLA Part 2, FSAR Subsection 14.2.3.1.5, from "...52.1 03g..." to "...52.103(g)..." in two places	Editorial
7423	STD,VCS	PT02	FSAR14	14.02.08	Revise COLA Part 2, FSAR Subsection 14.2.8, second inserted paragraph that begins "The sequential schedule... from "...SSC..." to "...structures, systems and components (SSC)..." and revise last sub-bullet under bullet Initial Test Program Schedule from "...Structures, Systems and Components (SSC)..." to "...SSC..."	Editorial
7646	STD,VCS	PT02	FSAR14	14.02.09.02.22	4. COLA Part 2, FSAR Chapter 14, new Subsection 14.2.9.2.22, will be added to read (with an LMA of STD COL 3.9-5): --- 14.2.9.2.22 Pressurizer Surge Line Testing (First Plant Only) Purpose The purpose of the pressurizer surge line testing is: a) to obtain data to verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line, and b) to obtain Reactor Coolant System piping displacement measurements for baseline data, as described in DCD subsections 3.9.3, 14.2.5, and 14.2.9.1.7 item (d). Prerequisites The construction tests for the individual components associated with the Reactor Coolant System have been completed. The testing and calibration of the required test instrumentation has been completed. The temporary sensors and instrumentation lead wires required for monitoring thermal stratification, cycling, and striping have been installed. The calibration of the transducers and the operability of the data acquisition equipment have been verified. Prior to testing of the piping system, a pretest walk-down shall be performed to verify that the anticipated piping movement is not obstructed by objects not designed to restrain the motion of the system (including instrumentation and branch lines). The system walk-down shall also verify that supports are set in accordance with the design. General Test Methods and Acceptance Criteria The performance of the Reactor Coolant System is observed and recorded during a series of individual tests that characterize the various modes of system operation. This testing verifies that the temperature sensors operate as described in DCD subsection 3.9.3 and in appropriate design specifications. a) Verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line. b) Record sensor data at specified intervals throughout hot functional testing of the RCS system, including during the drawing and collapsing of the bubble in the pressurizer. c) Retain the following plant parameters time history for the same data recording period: • Hot leg temperature • Reactor Coolant System pressure • Reactor coolant pump status • Pressurizer level	VEGP-RAI-LTR 057 response to RAI 03.12-002 item 4, SNC Ltr ND-10-1263

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					• Pressurizer temperature (liquid and steam) • Pressurizer spray temperature • Pressurizer spray and auxiliary spray flow • Normal residual heat removal system flow rate • Passive core cooling system – passive residual heat removal flow rate. d) Monitor pressurizer surge line and pressurizer spray line for valve leakage. e) Remove the transducers and associated hardware after the completion of testing. f) Proper operation of the temperature sensors in the pressurizer surge and spray lines is verified.	
7424	VCS	PT02	FSAR14	14.03.02.03.02	2. COLA Part 2, FSAR, Section 14.3.2.3.2, Physical Security ITAAC (PS-ITAAC), will be revised from: Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI) as outlined in Appendix C.II.I-C of Regulatory Guide 1.206. These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements. To Read: [Reviewer's Note: A new left-margin annotation (LMA) STD COL 13.6-1 will be applied to this paragraph. The current LMA, STD SUP 14.3-1, applies only to Subsection 14.3.2.3.3, Other Site-Specific Systems.] Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI). These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements.	RAI 13.6.1-13 (Letter 077 Response per SCE&G Letter NND-10-0236) VEGP RAI LTR 047 S2 response to RAI 14.03.12-001 item 2 (SNC Ltr ND-10-0886)
7427	STD,VCS	PT02	FSAR14	14.03.02.03.03	COLA Part 2, FSAR Subsection 14.3.2.3.3, first sentence, will be revised from: "No additional site-specific systems meet the ITAAC selection criteria." To read: "One additional site-specific system has been determined to meet the ITAAC selection criteria, and ITAAC have been included for the Transmission Switchyard and Offsite Power System (ZBS) as indicated in Table 14.3-201."	Rev 2 update of Table 14.3-201 per Qb 5256
7535	STD,VCS	PT02	FSAR14	14.03.03	Add the following information in Section 14.3.3 --- 14.3.3 CDM SECTION 3.0, NON-SYSTEM BASED DESIGN DESCRIPTIONS AND ITAAC --- Add the following new subsection after the first paragraph in DCD Subsection 14.3.3	Editorial to add Section Headings as a result of qB item 7426.
7718	STD,VCS	PT02	FSAR14	14.03.03	COLA Part 2, FSAR Chapter 14, Subsection 14.3.3.# (where # is the next sequential number), Piping Design ITAAC, LMA is revised from STD COL 3.9-1 to STD COL 3.9-7 (which is the COL item addressed in the Basis letter).	Revision to LMA for item in COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 5 SNC Letter ND-10-0801
7426	STD,VCS	PT02	FSAR14	14.03.03.01	5. COLA Part 2, FSAR Chapter 14, Subsection 14.3.3, add the following Subsections 14.3.3.# (where # is the next sequential number) and text, (note that the first item added will have an LMA of STD COL 3.6-1, and the second item added will have an LMA of STD COL 3.9-1) as follows: 14.3.3.1 Pipe Rupture Hazard Analysis ITAAC	COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 5 SNC Letter ND-10-0801

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					A pipe rupture hazard analysis is part of the piping design. The analyses will document that structures, systems, and components (SSCs) which are required to be functional during and following a design basis event have adequate high-energy and moderate-energy pipe break mitigation features. The locations of postulated ruptures and essential targets will be established and required pipe whip restraint and jet shield designs will be included. The as-designed pipe rupture hazards analysis will be based on the as-designed piping analysis and will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The report of the pipe rupture hazard analysis shall conclude that, for each postulated piping failure, the systems, structures, and components that are required to be functional during and following a design basis event are protected. The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 are covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built pipe rupture hazards mitigation features reflect the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed. The as-designed pipe rupture hazard analysis completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure. --- 14.3.3.2 Piping Design ITAAC The piping design ITAAC consists of the piping analysis for safety-related ASME Code piping. The piping design is completed on a package-by-package basis for applicable systems. In order to support closure of the piping design ITAAC, information consisting of the as-designed piping analysis for piping lines chosen to demonstrate all aspects of the piping design will be made available for NRC review, inspection, and/or audit. This information will consist of a design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class I piping. The piping packages to be analyzed are identified the in DCD. The ASME Code prescribes certain procedures and requirements that are to be followed for completing the piping design. The piping design ITAAC includes a verification of the ASME Code design report to ensure that the appropriate code design requirements for each system's safety class have been implemented. A reconciliation of the applicable safety-related as-built piping systems is covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built piping reflects the design, is reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed. The piping design completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure. ---	
7428	STD,VCS	PT02	FSAR14	14.04.04	Revise COLA Part 2, FSAR Subsection 14.4.4, from "...hold points is addressed..." to "...hold points are addressed..."	Editorial
PT02 - FSAR15						**1 COLA Change**
7429	STD,VCS	PT02	FSAR15	15.00	2. COLA Part 2, FSAR Chapter 15, Section 15.0, will be revised from: This section of the referenced DCD is incorporated by reference with no departures or supplements. To read (new sections will include LMA of STD COL 15.0-1): This section of the referenced DCD is incorporated by reference with the following departures and/or	DCD Rev 18, Based on WEC letter DCP/NRC2461 dated 20090506 COL-SER-OI-Ch15 response via ND-10-

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					supplements. --- 15.0.3.2 Initial Conditions --- Add the following paragraph at the end of DCD Subsection 15.0.3.2. The actual selected plant operating instrumentation has documented instrumentation uncertainties to calculate a primary power calorimetric uncertainty that confirms the uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric values. --- 15.0.15 Combined License Information --- Add the following text to the end of DCD Subsection 15.0.15.1. This COL item is addressed in FSAR Subsection 15.0.3.2. ---	1018
PT02 - FSAR17						**6 COLA Changes**
7554	VCS	PT02	FSAR17	17.01	In a future revision to the VCSNS COLA, the fifth paragraph of FSAR Subsection 17.1 will be revised to include a statement of conformance to Regulatory Guide 1.28 Revision 0 for the Unit 1 QA program: SCE&G maintains oversight of the COL application development as well as design and construction activities under its existing 10 CFR Part 50, Appendix B, program as described in the NRC approved SCE&G V.C. Summer Nuclear Station Unit 1 "Operational Quality Assurance Plan" (Reference 206). The Unit 1 Quality Assurance Program complies with the requirements of Regulatory Guide 1.28, Revision 0 - "Quality Assurance Program Requirements (Design and Construction)". The "Operational Quality Assurance Plan" (Reference 206) is supplemented, in part, by the SCE&G "New Nuclear Deployment Quality Assurance Plan" (Reference 204). The "New Nuclear Deployment Quality Assurance Plan (Reference 204) serves as an interfacing document between the work activities of the New Nuclear Deployment organization and the "Operational Quality Assurance Plan" (Reference 206). It assures that the proper administrative controls and the quality of activities related to the procurement of services, equipment, oversight of construction/ manufacturing, and licensing activities being petiormed within the New Nuclear Deployment organization conform to the applicable requirements of 10 CFR 50, Appendix B. These plans provide the necessary quality assurance guidance for oversight of site characterization activities and COL application content providers. SCE&G maintains this oversight through the review and approval of the NuStart Quality Assurance Plan (Reference 201) and industry standard COL application sections as well as conducting audits/surveillances of Bechtel activities, and providing input to the COL application development, including, but not limited to, review of COL application content.	RAI 17.5-7 S1 (SCE&G Letter NND-10-0143)
7431	STD,VCS	PT02	FSAR17	17.05	3. COLA Part 2, FSAR Chapter 17, Section 17.5, will be revised to read: The Quality Assurance Program in place during the design, construction, and operations phases is described in the QAPD, which is maintained as a separate document. This QAPD is incorporated by reference. This QAPD is based on NEI 06-14A, "Quality Assurance Program Description" (Reference 207)	COL-SER-OI-Ch17 S1 response to OI 17.01-001 item 3 SNC Letter ND-09-2082
7433	STD,VCS	PT02	FSAR17	17.05	1. COLA Part 2, FSAR Chapter 17, Section 17.5, will be revised in the first paragraph to add a STANDARD reference to Table 1.6-201 from (the LMAs remain unchanged): This QAPD is incorporated by reference. To read:	COL-SER-OI-Ch 17 response to OI 17.05-009 VR2 item 1 (SNC Ltr ND-10-0702)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					This QAPD is incorporated by reference (see Table 1.6-201).	
7577	VCS	PT02	FSAR17	17.05	Revise COLA Part 2, FSAR Section 17.5 to add left margin annotation (LMA) STD COL 17.5-4.	Editorial
7435	STD,VCS	PT02	FSAR17	17.06	COLA Part 2, FSAR Chapter 17, Section 17.6, will be revised to include the following new paragraph at the end of the section with a left margin annotation (LMA) of STD SUP 17.6-2: Condition monitoring of underground or inaccessible cables is incorporated into the maintenance rule program. The cable condition monitoring program incorporates lessons learned from industry operating experience, addresses regulatory guidance, and utilizes information from detailed design and procurement documents to determine the appropriate inspections, tests and monitoring criteria for underground and inaccessible cables within the scope of the maintenance rule (i.e., 10 CFR 50.65). The program takes into consideration Generic Letter 2007-01.	VEGP RAI LTR 053 response to RAI 08.02-014 SNC Ltr ND-10-0813
7436	STD,VCS	PT02	FSAR17	17.08	COLA Part 2, FSAR Chapter 17, Section 17.8, will be revised to read: 207. Nuclear Energy Institute, Technical Report NEI 06-14A, "Quality Assurance Program Description," Revision 7, July 2009.	COL-SER-OI-Ch17 S1 response to OI 17.01-001 item 4 SNC Letter ND-09-2082
PT02 - FSAR19						**6 COLA Changes**
7043	VCS	PT02	FSAR19	19.58	COLA Part 2, FSAR Chapter 19, Section 19.58 will be revised from: This section of the referenced DCD is incorporated by reference with no departures or supplements. To read: This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.	RAI 19-78 (Letter 071 Response per NND-09-0329)
7044	VCS	PT02	FSAR19	19.58	COLA Part 2, FSAR Chapter 19, Section 19.58 will be revised to add new Subsection 19.58.3 (with LMA VCS SUP 19.58-1) that reads: Table 19.58-201 documents the site specific external events evaluation that has been performed to VCSNS Units 2 and 3. This table provides a general explanation of the evaluation and resultant conclusions and provides a reference to applicable sections of the FSAR where more supporting information (including data used, methods and key assumptions) regarding the specific event is located. Based upon this evaluation, it is concluded that the VCSNS Units 2 and 3 site is bounded by the High Winds, Floods and Other External Events analysis documented in DCD Section 19.58 and APP-GW-GLR-101 (Reference 201) and no further evaluations are required at the COL application stage.	RAI 19-78 (Letter 071 Response per NND-09-0329)
7045	VCS	PT02	FSAR19	19.58	COLA Part 2, FSAR Chapter 19, Section 19.58 will be revised to add new Subsection 19.58.4 that reads: 19.58.4 References 201. Westinghouse Electric Company LLC, "AP1000 Probabilistic Risk Assessment Site-Specific Considerations," Document Number APP-GW-GLR-101, Revision 1, October 2007.	RAI 19-78 (Letter 071 Response per NND-09-0329)
7673	VCS	PT02	FSAR19	19.58	COLA Part 2, FSAR Chapter 19, Section 19.58 will be revised to add new Subsection 19.58.3 as follows: --- 19.58.3 CONCLUSION Add the following information at the end of DCD Subsection 19.58.3:	Addition of Subsection heading and lead-in sentence for incorporation of RAI 19-78 (Letter 071 Response per NND-09-0329)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					(insert QB Item 7044) ----------	
7046	VCS	PT02	FSAR19	19.58.T / T 19.58-201	COLA Part 2, FSAR Chapter 19, Section 19.58, add new Table 19.58-201 as shown in Attachment 1 to RAI 19.78.	RAI 19-78 (Letter 071 Response per NND-09-0329)
7215	VCS	PT02	FSAR19	19.59.10.05	COLA Part 2, FSAR Chapter 19, Subsection 19.59.10.5 will be revised from: "It has been confirmed that the Winds, Floods, and Other External Events analysis documented in DCD Section 19.58 is applicable ot the site. The site-specific design has been evaluated and is consistent with the AP1000 PRA assumptions. Therefore, Chapter 19 of the AP1000 DCD is applicable to this design." to read: "As discussed in Section 19.58.3, it has been confirmed that the Winds, Floods, and Other External Events analysis documented in DCD Section 19.58 is applicable ot the site. The site-specific design has been evaluated and is consistent with the AP1000 PRA assumptions. Therefore, Section 19.58 of the AP1000 DCD is applicable to this design."	RAI 19-78 (Letter 071 per NND-09-0329)
PT05 - *(empty)*						**7 COLA Changes**
7681	VCS	PT05		Annex 2, Page 2-9	Revise Page 2-9 from : Unit 2 EALs to be developed in accordance with NEI 07-01, Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors. To Read: Intentionally left blank	RAI 13.03-41 (Letter 062 Response per NND-09-0279)
7682	VCS	PT05		Annex 3, Page 3-9	Revise Page 3-9 from : Unit 2 EALs to be developed in accordance with NEI 07-01, Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors. To Read: Intentionally left blank	RAI 13.03-41 (Letter 062 Response per NND-09-0279)
7697	VCS	PT05		H.2	Revise Section H.2 to add the following new bullet to the EOF design considerations: "It is designed to support a remote TSC and remote OSC in the event of an emergency which limits access to the site."	RAI 13.03-50 (Letter 078 S1 Response per NND-10-0291)
7678	VCS	PT05		H.7.c	Revise Section H.7.c to remove statement regarding use of NRC Mobile Laboratories. Revise From: Outside analytical assistance may be requested from state and federal agencies, or through contracted vendors. The state maintains a radiological laboratory that provides independent analysis. The NRC mobile laboratory may be made available for Site Area Emergencies and General Emergencies. The DOE, through the Interagency Radiological Assistance Program has access to any national laboratory with DOE contract (i.e., Savannah River Site, Brookhaven, Oak Ridge, Lawrence Livermore, etc.).	RAI 13.03-46 (Letter 078 Response per NND-10-0142)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					To Read: Outside analytical assistance may be requested from state and federal agencies, or through contracted vendors. The state maintains a radiological laboratory that provides independent analysis. The DOE, through the Interagency Radiological Assistance Program has access to any national laboratory with DOE contract (i.e., Savannah River Site, Brookhaven, Oak Ridge, Lawrence Livermore, etc.).	
5870	VCS	PT05		K.05.0b	Revise Section K.5.b as follows: Contamination Control Means: Personnel found to be contaminated will normally be attended to at decontamination areas located onsite. Temporary decontamination areas can also be set up inside at various locations. Decontamination showers and supplies are provided onsite in the Health Physics area located in the Annex Building of the AP1000 units along with additional personnel decontamination equipment and capabilities. Basic decontamination supplies such as soaps, shampoo, mild detergent, 3% Hydrogen Peroxide solution, plastic bags, plastic suits, cotton swabs, oral hygiene products, and saline solution will be available in the Health Physics area. Shower and sink drains in the controlled area are routed to the miscellaneous waste processing system where the liquid is processed and monitored prior to discharge. Potentially contaminated emergency vehicles will be surveyed before they are allowed to leave the plant or offsite assembly area. If the survey area is not suitable for monitoring and decontamination due to radiological or other concerns, vehicles will be surveyed at an alternate location.	RAI 13.03-39 (Letter 062 Response per NND-09-0279)
7710	VCS	PT05		K.05.0b	Revise Section K.5.b as follows: Contamination Control Means: Personnel found to be contaminated will normally be attended to at decontamination areas located onsite. Temporary decontamination areas can also be set up inside at various locations. Decontamination showers and supplies are provided onsite in the Health Physics area located in the Annex Building of the AP1000 units along with additional personnel decontamination equipment and capabilities. Basic decontamination supplies such as soaps, shampoo, mild detergent, 3% Hydrogen Peroxide solution, plastic bags, plastic suits, cotton swabs, oral hygiene products, and saline solution will be available in the Health Physics area. Shower and sink drains in the radiologically controlled area are routed to the miscellaneous waste processing system where the liquid is processed and monitored prior to discharge. Potentially contaminated emergency vehicles will be surveyed before they are allowed to leave the plant or offsite assembly area. If the survey area is not suitable for monitoring and decontamination due to radiological or other concerns, vehicles will be surveyed at an alternate location.	Editorial Correction to RAI 13.03-39 (Letter 062 Response per NND-09-0279) to correct "radiological controlled area" to "radiologically controlled area"
7311	VCS	PT05		Section F	In Section F of the Emergency Plan, revise the description of the 800Mhz Radio System From: 800Mhz Radio: This radio is available as a backup notification device to the offsite authorities at selected county warning points. To Read: 800MHz Radio: This radio system is an 800 MHz SCANA Corporation system that is divided into trunks which are used by corporation subsidiaries. The trunk system at VCSNS is comprised of channels for Maintenance, Operations, Health Physics, Field Monitoring Teams, etc. to allow ameans of communications between facility personnel and field personnel for routine work and emergency conditions. The system utilizes both base stations and remote units in conjunction with associated cabling, repeaters, and antennas to provide optimum coverage for two-way continuous transmission.	RAI 09.05.02-8 (Letter 073 per NND-10-0104)
PT10 - *(empty)*						**16 COLA Changes**
7659	STD,VCS	PT10		LC #03	13. COLA Part 10, Proposed License Condition 3, "Operational Program Implementation," will be revised to add the following new milestone (where the # is replaced with the next appropriate number): D.# – Emergency Planning (applicable portions)	VEGP-COL-SER-OI-CH01 S2 response to OI 01.05-001 item 13 SNC Ltr ND-10-1305
7469	STD,VCS	PT10		LC#02, 03.06-1	6. COLA Part 10, Proposed License Conditions, item 2 – COL Item No. 3.6-1, will be revised from (Note that this	COL-SER-OI-Ch03 S6

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					revised item essentially identifies a milestone for advance completion of the ITAAC discussed in 14.3.3): 3.6-1 Pipe Break Hazards Analysis 3.6.4.1 Prior to initial fuel load After a Combined License is issued, the following activity will be completed by the COL holder: A pipe rupture hazard analysis is part of the piping design. It is used to identify postulated break locations and layout changes, support design, whip restraint design, and jet shield design. The final design for these activities will be completed prior to fabrication and installation of the piping and connected components. The as-built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load. To read: 3.6-1 As-Designed Pipe Rupture Hazards Analysis 3.6.4.1 Prior to installation of the piping and connected components in their final location After a Combined License is issued, the following activity will be completed by the COL holder. An as-designed pipe rupture hazard evaluation will be available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets and appropriate mitigation features (Reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate. A pipe rupture hazards analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components (SSCs), which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.	response to OI 03.06-001 item 6 SNC Letter ND-10-0801
7471	STD,VCS	PT10		LC#02, 03.09-2	7. COLA Part 10, Proposed License Conditions, item 2 – COL Item No. 3.9-2, will be deleted since this item is addressed by ITAAC in DCD Tier 1 Section 2 line items for the applicable systems.	COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 7 SNC Letter ND-10-0801
7476	STD,VCS	PT10		LC#02, 15.00-1	3. COLA Part 10, Proposed License Conditions, LC#2, will be revised to include a new line item for COL item 15.0-1 as follows: 15.0-1 Documentation of Plant Calorimetric Uncertainty Methodology 15.0.15.1 Prior to initial fuel load Confirm the plant operating instrumentation installed for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus™ System.	DCD Rev 18, Based on WEC letter DCP/NRC2461 dated 20090506 COL-SER-OI-Ch15 S1 response via ND-10-1018
7478	STD,VCS	PT10		LC#03	2. COLA Part 10, Proposed License Conditions (Including ITAAC), Section 3, 2nd paragraph will be revised from: PROPOSED LICENSE CONDITION: The licensee shall implement the programs or portions of programs identified below on or before the associated milestones identified below. A. Construction Initiation – The licensee shall implement each operational program identified below prior to initiating construction of nuclear safety- or security-related structures, systems, or components.	VEGP-RAI-LTR 049 S1 response to RAI 13.06-035 item 2 SNC Ltr ND-10-1230\ RAI 13.6.1-1 (Letter 076 S1 Response per NND-10-240)

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					A.1 – Fitness for Duty (Construction) B. 18 Months Prior to Fuel Load – The license shall implement each operational program identified below at least 18 months prior to scheduled date of initial fuel load. B.1 – Reactor Operator Training C. Receipt of Materials – The licensee shall implement each operational program identified below prior to initial receipt of byproduct, source, or special nuclear materials onsite (excluding Exempt Quantities as described in 10 CFR 30.18). C.1 – Radiation Protection (applicable portions) C.2 – Fire Protection Program (applicable portions) C.3 – Non Licensed Plant Staff Training Program (applicable portions) C.4 – Emergency Planning (applicable portions) C.5 – Security Program (applicable portions) D. Fuel Receipt – The licensee shall implement each operational program identified below prior to initial receipt of fuel onsite. D.1 – Fire Protection (applicable portions) D.2 – Radiation Protection (applicable portions) D.3 – Security Program (applicable portions) E. Construction Testing – The licensee shall implement each operational program identified below prior to initial construction testing. E.1 – Initial Test Program – Construction Testing F. Preoperational Testing – The licensee shall implement each operational program identified below prior to initial preoperational testing. F.1 – Initial Test Program – Preoperational Testing G. Fuel Loading – The licensee shall implement each operational program identified below prior to initial fuel load. G.1 – Environmental Qualification G.2 – Pre-Service Testing G.3 – Process and Effluent Monitoring and Sampling G.4 – Radiation Protection (applicable portions) G.5 – Motor-Operated Valve Testing G.6 – Fire Protection G.7 – Fitness for Duty (Operations) G.8 – Containment Leakage Rate Testing G.9 – Physical Security G.10 – Cyber Security To read: PROPOSED LICENSE CONDITION: The licensee shall implement the programs or portions of programs identified below on or before the associated milestones identified below.	

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					A. Construction Initiation – The licensee shall implement each operational program identified below prior to initiating construction of nuclear safety- or security-related structures, systems, or components. None identified. B. 18 Months Prior to Fuel Load – The license shall implement each operational program identified below at least 18 months prior to scheduled date of initial fuel load. B.1 – Reactor Operator Training C. Receipt of Materials – The licensee shall implement each operational program identified below prior to initial receipt of byproduct, source, or special nuclear materials onsite (excluding Exempt Quantities as described in 10 CFR 30.18). C.1 – Radiation Protection (applicable portions) C.2 – Fire Protection Program (applicable portions) C.3 – Non Licensed Plant Staff Training Program (applicable portions) C.4 – Emergency Planning (applicable portions) C.5 – Security Program (applicable portions) D. Fuel Receipt – The licensee shall implement each operational program identified below prior to initial receipt of fuel onsite. D.1 – Fire Protection (applicable portions) D.2 – Radiation Protection (applicable portions) D.3 – Security Program (applicable portions) E. Construction Testing – The licensee shall implement each operational program identified below prior to initial construction testing. E.1 – Initial Test Program – Construction Testing F. Preoperational Testing – The licensee shall implement each operational program identified below prior to initial preoperational testing. F.1 – Initial Test Program – Preoperational Testing G. Fuel Loading – The licensee shall implement each operational program identified below prior to initial fuel load. G.1 – Environmental Qualification G.2 – Pre-Service Testing G.3 – Process and Effluent Monitoring and Sampling G.4 – Radiation Protection (applicable portions) G.5 – Motor-Operated Valve Testing G.6 – Fire Protection G.7 – Deleted G.8 – Containment Leakage Rate Testing G.9 – Physical Security G.10 – Cyber Security	
7479	STD,VCS	PT10		LC#05	COLA Part 11, Cyber Security Plan, Section 3.1.1, Security Assessment and Authorization, will be revised from: 5. SECURITY PROGRAM REVISIONS: An implementation license condition approved in the SRM regarding SECY-05-0197 applies to the security	VEGP RAI LTR 51 Erratum 4 - SNC ND-10-0773

Change ID#	COLA REP	COLA Part REP	Chapter REP	Section/Page REP	Complete Change Description	Basis for Change
					program. PROPOSED LICENSE CONDITION: The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97, and Section VIII of Appendix D to Part 52 when nuclear fuel is on site, and continuing until all nuclear fuel is permanently removed from the site. To read: 5. SECURITY PROGRAM REVISIONS: An implementation license condition approved in the SRM regarding SECY-05-0197 applies to the security program. PROPOSED LICENSE CONDITION: The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, safeguards contingency plan, and cyber security plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97, and Section VIII of Appendix D to Part 52 when nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.	
7481	STD,VCS	PT10		LC#06	Associated VEGP COL Application Revisions: 1. Revise COL Application Part 10, License Conditions, proposed license condition 6, final sentence introduction to the list from: This schedule shall address: To read: This schedule shall also address:	COL-SER-OI-CH03 S2 response to OI 03.10-001 item 1 SNC Letter ND-10-0187
7483	STD,VCS	PT10		LC#06	2. COLA Part 10, Proposed License Conditions (including ITAAC), License Condition 6, Proposed License Condition for Operational Program Readiness, will be revised to add the following line item: #- the spent fuel rack Metamic coupon monitoring program implementation. (Note -# will be replaced with the next sequential number in the COLA)	COL-SER-OI-Ch09 S1 response to OI 09.01-001 item 2 SNC Letter ND-10-0781
7484	STD,VCS	PT10		LC#06	COLA Part 10, Proposed License Conditions (Including ITAAC), VEGP Proposed License Condition 6, Operational Program Readiness, will be revised from: 6. OPERATIONAL PROGRAM READINESS: The NRC inspection of operational programs will be the subject of the following license condition in accordance with SECY-05-0197. PROPOSED LICENSE CONDITION: The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall address: a. the implementation of site specific Severe Accident Management Guidance. b. the reactor vessel pressurized thermal shock evaluation at least 18 months prior to initial fuel load. c. the approved preoperational and startup test procedures in accordance with FSAR Subsection 14.2.3.	VEGP-RAI-LTR-054 response to RAI 19-95 SNC Ltr ND-10-1020

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					d. the flow-accelerated corrosion (FAC) program implementation, including the construction phase activities. To read: 6. OPERATIONAL PROGRAM READINESS: The NRC inspection of operational programs will be the subject of the following license condition in accordance with SECY-05-0197. PROPOSED LICENSE CONDITION: The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall address: a. the implementation of site specific Severe Accident Management Guidance. b. the reactor vessel pressurized thermal shock evaluation at least 18 months prior to initial fuel load. c. the approved preoperational and startup test procedures in accordance with FSAR Subsection 14.2.3. d. the flow accelerated corrosion (FAC) program implementation, including the construction phase activities. e. full implementation of the operational and programmatic elements of responding to an event associated with a loss of large areas of the plant due to explosions or fire, prior to initial fuel load.	
7485	STD,VCS	PT10		LC#07	COLA Part 10, Proposed License Conditions, including ITAAC, will be revised to include a new License condition. Line item 7 will be revised from: 7. Not Used To read: 7. First-Plant-Only and First-Three-Plant-Only Testing Certain design features of the AP1000 plant will be subjected to special tests to establish unique phenomenological performance parameters of the AP1000 design. Because of the standardization of the AP1000 design, these special tests (designated as first-plant-only tests and first-three-plant-only tests) are not required on subsequent plants. These tests will be controlled through license conditions to ensure that relevant test results are reviewed, evaluated, and approved by the designated licensee management before proceeding with the next testing phase. Accordingly, the following license condition is proposed: First-Plant-Only and First-Three-Plant-Only Testing Following completion of the testing, the licensee completing the testing shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and tests, as required, are performed. Additionally, the licensee completing the testing shall also provide written notification of completion of the testing to the Director of the Office of New Reactors. 1. For testing completed during pre-critical testing, criticality testing, and during low-power testing, these reports may be in conjunction with the test completion reports required for the power ascension test phase as	VEGP-RAI-LTR 058 response to RAI 14.02-001 SNC Ltr ND-10-1202

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					identified below. 2. For tests completed during operation above 5% RTP, the reports shall be provided for each individual test within thirty (30) calendar days of the licensee confirmation of completion of the testing. Subsequent plant licensees crediting completion of testing by the first-plant or by the first-three-plants shall provide a report referencing the written notification of completion submitted by the plant(s) completing the testing to the Director of the Office of New Reactors.	
7486	STD,VCS	PT10		LC#09	COLA Part 10, Proposed License Conditions, including ITAAC, will be revised to include a new license condition. Line item 9 will be revised from: 9. Not Used To read: 9. Power-Ascension Test Phase Certain milestones within the startup testing phase of the initial test program (i.e., pre-critical testing, criticality testing, and low-power (<5% RTP) testing) are controlled through license conditions to ensure that relevant test results are reviewed, evaluated, and approved by the designated licensee management before proceeding with the power ascension test phase. Accordingly, the following license conditions are proposed: Pre-critical and Criticality Testing 1. Following completion of pre-critical and criticality testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed. 2. The licensee shall provide written notification to the Director of the Office of New Reactors within fourteen (14) calendar days of completion of the pre-critical and criticality testing. Low-Power (<5% RTP) Testing 1. Following completion of low-power (<5% RTP) testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed. 2. The licensee shall provide written notification to the Director of the Office of New Reactors within fourteen (14) calendar days of completion of the low power testing.	VEGP-RAI-LTR 059 response to RAI 14.02-002 SNC Ltr ND-10-1203
7489	STD,VCS	PT10		LC#AppB B2 piping	9. Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, add the following after the last site-specific ITAAC (where # is the next sequential number): Pipe Rupture Hazard Analysis ITAAC The ITAAC for Pipe Rupture Hazard Analysis are included in attached Table 3.8-#. Piping Design ITAAC The ITAAC for Piping Design are included in attached Table 3.8-#.	COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 9 SNC Letter ND-10-0801
7663	VCS	PT10		LC#AppB B2 piping	Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, add the following lead-in statement for non-system based site specific ITAAC before the Emergency Planning ITAAC heading: "The following non-system based site specific ITAAC are provided:"	Addition of new non-system ITAAC section

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7490	STD,VCS	PT10		LC#AppB B3 Security	3. COLA Part 10, Proposed License Conditions (Including ITAAC), Appendix B, Inspections, Tests, Analysis, and Acceptance Criteria, Table 2.6.9-2 - SITE-SPECIFIC PHYSICAL SECURITY INSPECTIONS, TESTS, ANALYSES AND ACCEPTANCE CRITERIA as noted in the final response letter	RAI 13.6.1-13 (Letter 077 S1 Response per NND-10-0236) VEGP RAI LTR 047 S2 response to RAI 14.03.12-001 item 3 SNC Ltr ND-10-0886
7492	STD,VCS	PT10		LC#AppB B6 piping	10. Part 10, Appendix B, Inspections, Tests, Analyses and Acceptance Criteria, add the following Tables 3.8-# Pipe Rupture Hazards Analysis (Sheet 1 of 1) and 3.8-# Piping Design (Sheet 1 of 1) after the last site-specific ITAAC Table. Refer to the final SER OI response letter for the complete tables.	COL-SER-OI-Ch03 S6 response to OI 03.06-001 item 10 SNC Letter ND-10-0801
7680	VCS	PT10		Table 3.8-1, Item 8	Revise Table 3.8-1, Item 8 to incorporate updated EP-ITAAC Acceptance Criteria as shown in NND-10-0142.	RAI 13.03-45 (Letter 078 Response per NND-10-0142)

SUMMARY

COLA Part REP	Chapter REP	Number of COLA Changes
PT01	*(empty)*	1
PT02	FSAR01	38
PT02	FSAR02	27
PT02	FSAR03	26
PT02	FSAR05	4
PT02	FSAR06	11
PT02	FSAR07	7
PT02	FSAR08	3
PT02	FSAR09	14
PT02	FSAR11	4
PT02	FSAR12	4
PT02	FSAR13	25
PT02	FSAR14	16
PT02	FSAR15	1
PT02	FSAR17	6
PT02	FSAR19	6
PT05	*(empty)*	7
PT10	*(empty)*	16
TOTALS (18 groups)		**216**