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A SCANA COMPANY

July 26, 2012
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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3
Combined License Nos. NPF-93 and NPF-94
Docket Nos. 52-027 and 52-028

Subject: Reporting of 10 CFR 50.59 Changes, Tests, and Experiments and 10 CFR 52 Appendix D Section VIII Departures

Reference: 1. Letter from Ronald B. Clary (SCE&G) to Document Control Desk (NRC), January 30, 2012 Update of Combined License Application Departure Report

In accordance with 10 CFR 50.59(d)(2), VCSNS Units 2 and 3 is required to submit a report to the NRC containing a brief description of any changes, tests or experiments made pursuant to 10 CFR 50.59(c), including a summary of the evaluation of each. This 10 CFR 50.59 report is for the period beginning January 30, 2012 and ending July 24, 2012. During that period there were no changes, tests or experiments made pursuant to paragraph (c) of 10 CFR 50.59.

Additionally, as required by paragraphs X.B.1 and X.B.3.b of Appendix D to 10 CFR Part 52, this submittal contains a report of all plant-specific departures made in this reporting period. The 10 CFR 52 Appendix D Departure Report is provided in Enclosure 1 to this letter and covers the period beginning in January 30, 2012 and ending July 24, 2012.

Should you have any questions, please contact Mr. Alfred M. Paglia by telephone at (803) 941-9876, or by email at apaglia@scana.com.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 26th day of July, 2012.

Sincerely,

Ronald A. Jones
Vice President
New Nuclear Operations

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NRO

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Enclosure 1: V.C. Summer Nuclear Station Units 2 and 3 Departure Report: January 30, 2012 through July 24, 2012

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V.C. Summer Nuclear Station Units 2 and 3 Departure Report

January 30, 2012 through July 24, 2012

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-002	Design finalization of structural modules, including the containment internal structures, identified that for many locations overlay plates, embedments, or back up structures are needed to satisfy criteria for the attachment of the supports and similar attachments to the liner plates of the steel plate concrete filled composite structures. The design of the CA01, CA02, CA05, and CA20 structural modules is changed to use ASTM A572 steel for liner plates in lieu of ASTM A36 steel. The higher strength liner plates will permit attachments at some locations without overlay plates. The portions of the module that use Duplex steel plates for corrosion resistance are not changed. The design of the spacing of the shear studs is changed to a 6 inch by 6 inch spacing for ASTM A572 liner plates. The requirements of American Institute of Steel Construction (AISC) N690-1994 continue to apply to the attachment design with the higher strength plates. The geometric configuration of the containment internals and walls in the Auxiliary Building are not changed.	This activity changes plate material for containment internal modules and modules in the auxiliary building. The change in the module design and resultant change in shear stud spacing satisfy the requirements and acceptance criteria in AISC N-690-1994 and DCD, Section 3.8.3.1.3 as did the original design. The geometric configuration, thickness, and strength of these structures are not adversely affected. There is no change in the design, analysis, or operation of the RCS or other plant systems. Based on the 10 CFR 50.59/10 CFR 52 Appendix D Section VIII screening of this change, prior NRC approval of the change is not required.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-004	DCD Subsection 9.3.5.2.2 is being corrected to present the plant design and be consistent with DCD Subsection 9.3.5.1.2. The DCD contradicts itself in that Subsection 9.3.5.2.2 incorrectly states that each sump is fitted with a vent connection to exhaust potential sump gases into the Radiologically Controlled Area Ventilation System (VAS) exhaust system. The VAS is a ventilation system in the Auxiliary and Annex Buildings. The liquid radwaste system (WLS), as described in DCD Subsection 9.3.5.1.2 accurately described the venting as the radioactive sump vents are directed to the ventilation system exhaust ducts serving the areas where the sump is located and that the containment sump vents directly to the containment. This activity corrects the Radioactive Waste Drain System (WRS) sump venting described in the DCD in that the containment sump is vented to containment rather than the VAS.	The plant equipment and design intent and philosophy have not changed. A contradiction in the DCD has been eliminated regarding a generalized statement about the WLS and WRS sump vents' repository. The AP1000 was designed with the correct vent philosophy provided in DCD Subsection 9.3.5.1.2, so there is no impact on SSCs. The plant design has not changed, but the DCD is being clarified regarding a generalized statement about the nonsafety-related WLS and WRS sump venting. There is no affect on structural analysis and the rewording in the DCD does not impact the Aircraft Impact Assessment. The change does not impact security barriers or radiation, protection and shielding safety analyses, nor does the change affect any procedure, method of evaluation, or test and experiment. The physical design of the sump vents has not changed, so there is no impact on ex-vessel severe accident consequences, containment venting and containment integrity. The VAS supply and exhaust ducts that ventilate the middle annulus are not affected by this departure and continue to be designed to be isolated for holdup and deposition of containment radioactive releases during a severe accident as discussed in the AP1000 Probabilistic Risk Assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-005	Design details and descriptions, as stated in DCD Revision 19, contained implied and stated fabrication and construction details including weld seams, course elevations, plate geometry, and attachments for the containment air baffle for the Containment Vessel. DCD Tier 2 Sections 3.8.2.1.1, 3.8.2.6, and 3.8.4.1.3, and DCD Figures 3.8.2-1 Sheet 1 and 3.8.4-1 Sheet 1 are revised to remove details regarding the fabrication and erection of the Containment Vessel. These details and figures are not intended to show required design and fabrication details. These changes are necessary to ensure that the DCD description is consistent with actual design and fabrication methods. Design details and descriptions, included in DCD Revision 19, provide fabrication and construction details (e.g., size and number of panels and detail design of supports and attachment) for the Containment air baffle and are unnecessary detail in the DCD. DCD Tier 2 Section 3.8.4.1.3 and DCD Figure 3.8.4-1 Sheet 1 are revised to remove details regarding the fabrication and construction of the containment air baffle. These details and figures are not intended to show required design and fabrication details. These details are inconsistent with the design finalization of the baffle and the fabrication details of the baffle, baffle panels, and supports.	The removal of the design and fabrication details does not adversely affect the containment vessel and containment air baffle design functions. It does not affect the method of performing or controlling design functions, nor does it have an effect on an evaluation for demonstrating that intended design functions will be accomplished. It removes detailed DCD information that is inconsistent with design and fabrication details. The removal of fabrication and construction details does not impact the design function of any SCC. The pressure retention and structural integrity function of the containment vessel is not adversely affected. The containment air baffle design function of providing for an air flow path for the passive containment cooling system is not adversely affected. The containment vessel design function to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents is not adversely affected. The facility is not being adversely changed by this activity. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-006	This activity is being made to enhance the functionality of containment sump level instrumentation post-Safe Shutdown Earthquake (SSE). Prior to this change, there was an inconsistency between the Reactor Coolant Pressure Boundary (RCPB) leak detection functionality and seismic classification and the DCD design requirements. Specifically, the containment sump module (KQ11), three containment sump level instruments (WLS-LT-034, WLS-LT-035, WLS-LT-036), and Primary Sampling System (PSS) radiation particulate monitoring instruments require modifications to comply with the current licensing commitments regarding plant operation post-SSE. Containment sump level monitoring, through the containment sump level instruments, is clarified to be the primary method of RCPB leakage detection in containment after an SSE. It provides conformance to position 6 of Regulatory Guide 1.45, although using different technology than envisioned in that guidance (sump level rather than airborne radioactivity). The containment sump level instruments indication in the main control room display remains non-seismic; however, SC-I local readout of the instruments is provided outside of containment and is qualified to be operable post-SSE. The Containment Atmosphere Radioactivity Monitor 18F particulate monitor remains seismic Category I, but the remaining tubing is not seismically qualified. This leakage detection system can be reasonably expected to remain functional following seismic events of lesser severity than the SSE; however, no special qualification program is used to assure operability under such conditions and no credit is taken for its functionality. It is clarified that the Containment Atmosphere Radioactivity Monitor is not the instrument used to provide RCPB leakage detection following seismic events that do not require plant shutdown in conformance to the intent of position 6 of Regulatory Guide 1.45; conformance to this position is provided by the containment sump level via the seismic Category I Containment Sump Level Monitoring system.	By enhancing the functionality of the containment sump level instrumentation post-SSE, the Reactor Coolant Pressure Boundary leakage detection function is unchanged. There is no affect on structural analysis and no impact on the Aircraft Impact Assessment. Additionally, enhancing the functionality of the containment sump level instrumentation post-SSE does not impact security barriers or radiation, protection and shielding safety analyses. These changes do not affect any procedure, method of evaluation or test and experiment. RCPB leakage detection instrumentation is not credited in the ex-vessel severe accident assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-008	This activity updates admixtures used in the production of concrete structures and modules described in the DCD. Admixtures are used to obtain certain concrete characteristics that would not be obtainable with a plain mix. The types of concrete admixtures are being revised to account for technology improvements that will allow for the production of conventional concrete and Self- Consolidating Concrete (SCC). Type B, C and F admixtures are added, Type D and vinsol admixtures are removed and Type A admixture use is clarified in the DCD. These changes are consistent with ASTM C494, ACI 349, and ACI 237R.	By allowing the use of admixture types B, C, and F and preventing the use of type D and vinsol, the concrete's design function is unchanged. The use of Self-Consolidating Concrete has no affect on structural analysis and the admixtures do not impact the Aircraft Impact Assessment. There is no adverse impact on concrete parameters such as strength, density, and durability. Additionally, these admixtures do not impact security barriers or radiation, protection and shielding safety analyses. These changes do not affect any procedure, method of evaluation, or test and experiment. The changes do not have an impact on ex-vessel severe accident consequences and do not impact core concrete interactions or containment pressurization due to core concrete interactions. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.
LCE-12-016	The principal construction code of the WGS Gas Cooler is categorized as ASME VIII/TEMA in DCD Rev. 19 Tier 2 Table 3.2-3. As a result of a previous design change, the WGS Gas Cooler was changed from a shell and tube heat exchanger to an off-the-shelf, dual tube coil heat exchanger. When this design change was incorporated into the DCD, Tier 2 Table 3.2-3 was not updated to reflect the principal construction code of the new heat exchanger. The correct principal construction code for the new heat exchanger is "Manufacturer Std."	This change involves modifying DCD Tier 2 Table 3.2-3 to accurately reflect the principal construction code of the WGS Gas Cooler. The design function of the WGS remains unchanged and the quality and construction/quality standards are not adversely affected. Therefore, this change does not adversely impact the design function of the WGS. This change does not affect any procedure, method of evaluation, or test and experiment. This activity does not impact a design feature credited in the ex-vessel severe accident assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.
LCE-12-022	This departure makes changes in the turbine building to the El. 82'-9" basemat area, concrete base pads, general layout arrangement, and various plant-specific DCD text changes for consistency with the Condensate Polishing System (CPS) resin rinse effluent design function. The turbine building El. 82'-9" basemat area is expanded north of column line 18 and south of column line 13.1. The concrete base pads that support structural columns 14, 15, 16, and 17 are lowered from El. 100'-0" to El. 90'-0", and a ditch has been created in the middle of the base pad for column 17. Stairwell S09 is removed, a new material handling	Implementing these changes has no adverse effect on structural analysis. The changes do not impact security barriers or radiation, protection and shielding safety analyses, nor does the change affect any procedure, method of evaluation, or test and experiment. There is no impact to ex-vessel severe accident consequences, containment venting, and containment integrity. The design functions of the turbine building and its structures, systems, and components as described in the plant-specific DCD or UFSAR continue to be met. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is

SCE&G Evaluation	Activity Description	Summary of Evaluation
	elevator is added, and stairwell S11 is relocated to the Northeast corner of El. 82'-9". Various DCD text changes are made to account for CPS resin rinse effluent being discharged to the turbine building sumps, and an additional pump is added to each sump to account for the additional volume and prevent overflowing.	required.
LCE-12-023	Detailed figures were provided in DCD Revision 19. This activity substitutes 46 DCD piping & instrumentation diagrams with simplified schematics such that all required information is maintained. It has been verified that the simplified figures together with associated DCD and FSAR text continue to provide sufficient understanding of design bases, safety analyses and facility operation. There is no change to the system design described in the DCD figures or supporting analysis. The actual system piping and instrumentation diagrams are not altered by this activity. This is a change to the level of detail documented in the DCD. The figure simplification effort removes extraneous detail from DCD figures.	There is no design function related to replacing existing DCD figures with simplified figures. This simplification effort does not impact the design function of any SSC. The actual system piping and instrumentation diagrams are not altered by this activity. This activity only simplifies DCD figures; no new design changes are proposed. The facility is not being changed by this activity. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-024	Update and clarify requirements for structural steel fabrication in structural steel fabrication and erection specifications APP-SS01-Z0-001 Revision 2 and APP-SS01-Z0-002 Revision 2. The changes apply to seismic Category I seismic Category II and non-seismic (see DCD Section 3.2.1). APP-GW-G1X-001, the Governing Codes & Standards Document is updated to be consistent with the changes in the steel fabrication specification and the DCD. DCD Sections 3.8.3.2 and 3.8.4.2 are updated to remove codes and standards that are referenced in the top level (parent) structural design codes. These are ACI-349-01, or AISC-N690-1994. The codes and standards removed are related to welding procedures and concrete specifications. DCD Sections 3.8.3.2 and 3.8.4.2 are updated to remove the revisions or dates for standards and specifications related to the detailing, placement, and specification of concrete. These standards and specifications do not include design or analysis requirements. Reference to the NCIC weld acceptance criteria is removed from DCD Sections 3.8.3.2 and 3.8.4.2 since it is referenced by the top level codes. The top level structural design codes (ACI-349-01 and AISC N-690-1994) are identified as Tier 2* information in these DCD sections and are not changed. Construction and fabrication requirements for seismic Category II structures are removed from DCD Section 3.7.2 since this section is about seismic analysis and not construction requirements. The seismic interaction between seismic Category I and seismic Category II structures are covered in Section 3.7.2.8 and are not changed.	The fabrication specification for structural steel, the DCD, and Governing Codes & Standards documents need clarification of the versions of codes and standards used for fabrication and installation of civil/structural commodities and structures. The codes and standards specified in the DCD and APP-GW-G1X-001 and the daughter standards and specifications cited in these top level codes and standards can provide multiple versions of the standards and specifications for fabrication. The changes to the fabrication specifications and the DCD clarify the standards and specifications to use. As a result of advances in industry standard practices and material manufacture, more recent versions of the standards and specifications should be specified for the purposes of fabrication and construction. Clarification of the requirements for fabrication and construction of steel structures does not change the design, analysis, or configuration of the AP1000 Seismic Category I and Seismic Category II structures. There is no adverse effect on the design function of these structures. The clarification of the requirements for fabrication and construction of steel structures has no impact on the procedures used to operate and control the AP1000 plant. The clarification of the requirements for fabrication and construction of steel structures has no impact on the design, analysis, and acceptance criteria for the AP1000 structures. The clarification of the requirements for fabrication and construction of steel structures does not require testing or an experiment. The clarification of the requirements for fabrication and construction of steel structures does not alter the response of systems, structure, and components in the AP1000 to an ex-vessel severe accident. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

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LCE-12-025	To complete integration of the FSAR and DCD, editorial changes are required to be made to the Plant-Specific DCD to ensure that the document continues to read consistently.	This activity is editorial, but does involve changes to information in the Plant-Specific DCD. Because of the editorial nature of the activity, no changes are being made to any descriptions of design functions, procedures, methodologies, tests, or experiments. Therefore, because this does not change any technical information, the change is determined to not require prior NRC approval in accordance with 10 CFR 52 Appendix D Section VIII.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-026	The change activity clarifies requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear Island structures. The information clarified includes test age of concrete, conformance with ACI standards, aggregate testing, use of air entraining admixtures, incorporation of waterstops, and ASTM specification tabulated.	The clarification of requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear island structures will not have an adverse impact on the strength of the nuclear island structures or the response of the structure to internal and external loads, including seismic loads. The nuclear island structures, with the clarification of requirements and commitments in the licensing basis, remains in compliance with ACI-349. The clarification of requirements and commitments in the licensing basis has no impact on design, analysis, or operation of safety related systems and components. The clarification of requirements and commitments in the licensing basis has no impact on plant operating procedures or on the control of the reactions in the core. The clarification of requirements and commitments in the licensing basis has no impact on the finite element analysis methods used to analyze the nuclear island structures. The analysis of the reactor coolant system and core to normal operation and postulated accident conditions is not impacted by the clarification of requirements and commitments in the licensing basis. The clarification of requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear island structures does not alter the assumptions or results of the ex-vessel severe accident assessment. The clarification of the requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear Island structures does not result in modification, addition to, or removal of a structure, system, or component (SSC) such that a design function is adversely affected, has no impact on plant operating procedures or on the control of the reactions in the core design function, does not result in an adverse change to a method of evaluation or use of an alternate method of evaluation, does not represent a tests or experiments outside the reference bounds of the design basis, and does not alter the assumptions or results of the ex-vessel severe accident assessment.

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LCE-12-027	This activity removes unused acronyms from the Plant-Specific DCD Table 1.1-1. The table contains approximately 17 acronyms which are identified in the table but which are unused throughout the Plant-Specific DCD and FSAR.	As this change does not involve a change to any underlying technical information and is solely a change to the UFSAR List of Acronyms, the activity does not adversely impact any design function or procedure, change a methodology, involve a test or experiment, or affect any EVSA feature. Because of this, the evaluation determined prior NRC approval was not required.
LCE-12-029	The change activity clarifies and revises details of the description of the basemat reinforcement design in the licensing basis for the nuclear island basemat. The information clarified addresses inconsistencies internal to a DCD figure and inconsistencies with the concrete dimensions. The rearrangement of the reinforcements is consistent with the design finalization.	The revision of the reinforcement arrangement in the licensing basis for the nuclear island basemat will not have an adverse impact on the strength of the nuclear island structures or the response of the structure to internal and external loads, including seismic loads. The nuclear island structures, with the change of reinforcement arrangement, remains in compliance with ACI-349. The ACI-349 requirements and criteria for the reinforcement provided to resist tension, flexure, and shear loads are satisfied with the revised arrangement. The clarification of requirements of the reinforcement design in the licensing basis and revision of reinforcement arrangement for the nuclear island basemat has no impact on design, analysis, or operation of safety related systems and components. The clarification of requirements and commitments in the licensing basis and revision of reinforcement arrangement has no impact on plant operating procedures or on the control of the reactions in the core. The clarification of the reinforcement design in the licensing basis and revision of reinforcement arrangement for the nuclear island basemat has no impact on the finite element analysis methods used to analyze the nuclear island structures. The analysis of the reactor coolant system and core to normal operation and postulated accident conditions is not impacted by the clarification of the reinforcement design in the licensing basis and revision of details for the nuclear island basemat. The clarification of the reinforcement design in the licensing basis and revision of reinforcement arrangement for the nuclear island basemat does not alter the assumptions or results of the ex-vessel severe accident assessment.

SCE&G Evaluation	Activity Description	Summary of Evaluation
LCE-12-030	The departure adds three vent (V114, V115A, V115B) and two drain (V116A, V116B) lines to DCD Tier 2 Figure 9.1-6 for the Spent Fuel Pool Cooling System (SFS), and a Normal Residual Heat Removal System (RNS) drain line (V065) to Tier 2 Tables 3.2-3, 3.11-1 and 31.6-3 and Tier 2 Figure 5.4-7. As part of the design finalization process, vents and drains are provided for the RNS and SFS. The vents and drains are placed on their associated system engineering drawing, added to the associated DCD Tier 2 SFS and RNS figures, and the (new) RNS drain line is added to three Tier 2 tables.	Within the licensing basis, the departure adds some additional details (i.e., vents and drains) to the Tier 2 figures for the SFS and RNS, and the new RNS drain line valve is added to three Tier 2 tables. The new vents and drains are added to allow for maintenance and system fill prior to system operation. The vents and drains are closed and capped during system operation, and thus, no system design function is adversely affected. No procedure, method of control, test or experiment is involved. The changes do not affect a defense-in-depth (Le., beyond design basis) function. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.