Thomas D. Gatlin
Vice President, Nuclear Operations
(803) 345-4342

SCE&G®
A SCANA COMPANY

December 31, 2013

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

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12
10 CFR 50.59 BIENNIAL REPORT

South Carolina Electric & Gas Company (SCE&G) hereby submits the Twenty-Third VCSNS Report pursuant to 10 CFR 50.59(d)(2).

This report contains a brief description and summary of the evaluations performed to support the changes and modifications made to the facility in accordance with 10 CFR 50.59(c) (Attachment). This report covers the time frame from October 1, 2011 to October 1, 2013.

If you have any questions or require additional information, please contact Bruce Thompson at (803) 931-5042.

Very truly yours,

Wayne D Stuart for

Thomas D. Gatlin

WCM/TDG/wt

Attachment - 10 CFR 50.59 Summary of Evaluations and Changes

c: K. B. Marsh
S. A. Byrne
J. B. Archie
N. S. Carns
J. H. Hamilton
J. W. Williams
W. M. Cherry
V. M. McCree
S. A. Williams
NRC Resident Inspector
K.M. Sutton
NSRC
RTS (LTD-324)
File (818.02-8, RR 8450)
PRSF (RC-13-0170)

IE47
NRR

10 CFR 50.59 Summary of Changes and Evaluations

Parent Document	Change Description	Evaluation Summary
NC (CR)-11-03227, Accumulator Pressure Monitoring for XVG01611A, B, C	Provide temporary plant computer monitoring of XVG01611A,B,C valve operator accumulator pressure indication, by disconnecting IPCS inputs from a feedwater temperature indicator (ITE03323) and two containment dew point monitors.	Plant operation under normal and accident conditions is not significantly affected by ITE03320 indication not being available to plant operators. The dewpoint monitors are only used during containment integrated leak rate testing. Temporarily disabling 2 of the dew point monitors does not affect this function because containment integrated leak rate testing is not performed during this time.
ECR-50780C, Alternate Seal Injection	The Alternate Seal Injection (ASI) provides an alternate means for Reactor Coolant Pump (RCP) seal injection to reduce the probability of RCP seal Loss of Coolant accident (LOCA).	The ASI system design, material and construction standards meet those of the structures, systems and components (SSCs) with which the ASI system interfaces. The power supply does not interface with the onsite vital power supply. The instrumentation and control is completely separate and independent. Actuation of the ASI system does not start or stop any other equipment. No existing onsite controls can start the ASI system. The Main Control Board is provided with five associated alarms including ASI diesel and ASI pump run alarms. The Main Control Board cannot remotely operate the system. There is no potential for control room operator error. Inadvertent actuation of the ASI System results in a slow transient which functionally represents a slow plant shutdown. The addition of cold, borated water to the reactor coolant system (RCS) is a benefit for departure from nucleate boiling (DNB): consequently there are no adverse impacts on fission product barriers. The method of analysis used an approved code.

10 CFR 50.59 Summary of Changes and Evaluations

Parent Document	Change Description	Evaluation Summary
ECR-50777, Online Breaching of Fire/Pressure Barriers Between Turbine Building and Control Building	This engineering change request (ECR) implements Bus 3 upgrades that require cable pulls from the Turbine Building to the cable spreading room. These specific maintenance activities require the opening of cable tray penetration between the Turbine Building and Control Building while the plant is operating.	The evaluation identified negligible impacts on the 425' Cable Spreading Room relative to pressure, temperature and relative humidity during a main steam line break in the Turbine Building.

ECR-50846, Weld Repair Contingency for Reactor Vessel (RV) Head Inspections (RF-20)
 	
This activity was approved by NRC Relief Request RR-III-09.
	The activity covers the repair of defects found during UT inspections of the RV head (per N-729-1) for RF-20. ECR-50846 is the implementing document for that repair in the event that repair is required.	The method of welding described in Westinghouse Commercial Atomic Power (WCAP)-15987 Revision 2 has been approved for use at Westinghouse plants per an Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) approved in December 2003, provided that the plant fulfills the criteria for use as defined in the WCAP. There are 2 applicable criteria for a plant to use this WCAP.
		
		1. The plant be of Westinghouse or Combustion Engineering design.
		    a. VC Summer is a 3-loop Westinghouse NSSS Reactor and thus meets this condition.
		
		2. Failure Effects Analysis (FEA) of the found flaws must support the ability to use the Westinghouse repair process.
		    a. This analysis shall be completed before the initiation of welding. Analysis shall support the use of WCAP-15987 Revision 2 for repair.
		
		Because VC Summer meets these conditions and the repair method has been

10 CFR 50.59 Summary of Changes and Evaluations

Parent Document	Change Description	Evaluation Summary
		approved by NRC, the proposed repair activity may be implemented without obtaining a License Amendment.
ECR-50585W & X, Safety-Related Chiller Replacement Activities	ECRs 50585W and 50585X replace the existing 330 ton nuclear safety related (NSR) Chillers with modern 281 ton chillers which reflect the current system design heat load. The modification provides for replacement of each of the safety related chillers, one at a time during plant operation so that both trains of Service Water and Chilled Water are available to support plant operation.	The modification made as a result of this activity conform fully to the current licensing basis for the plant. All functions described in FSAR section 9.4.7.2.4 will be performed identically by the replacement chillers. The redesign of the new chillers to operate down to a lower load eliminates the need for operator action to throttle Service Water. The elimination of this manual action to throttle is a slight change, but is an improvement, and is acceptable under the current licensing basis. The existing chiller controls use analog controllers and relay logics, whereas, following this modification, the control system for each of these chillers will be a Triconex PLC Class 1E digital controller. The detailed evaluation in the following sections has been performed in accordance with the NRC-endorsed NEI 01-01 guidelines, demonstrates that: a) The Triconex digital control system Topical Report 7286-545-1-A has been accepted by NRC in its SER dated December 12, 2001 for safety-related use in nuclear power plants; b) All plant-specific conditions of approvals specified in the NRC SER have been satisfied for the replacement chiller application, as discussed in detail in Appendix 2A; c) The potential for software common mode failure has been carefully considered. The Triconex plans,

10 CFR 50.59 Summary of Changes and Evaluations

Parent Document	Change Description	Evaluation Summary
		procedures, quality assurance (QA) and verification and validation (V&V) for the Triconex operating system software are extensive and robust, and have been approved in the NRC SER. The application software has been developed by Nuclear Logistics Incorporated (NLI) under its plans, procedures, QA and V&V, and tested extensively at the factory. On this basis, the potential for a software common mode failure that could disable all chillers is concluded to be highly unlikely, no more likely than the potential for common cause failures in the original chiller equipment; and d) This activity may be performed under 10CFR50.59.
NC (CR)-13-01463, Incomplete Zirlo Coil Annealing	Westinghouse notified VC Summer of a non-conforming condition where some portion of Zirlo coil material received an incomplete annealing condition.	The presence of insufficiently annealed grid strap material was identified at a Westinghouse sub-supplier, and the Westinghouse Columbia Fuel Fabrication Facility. This material originated at the Westinghouse Western Zirconium site. Based on the current extent of condition, this material is potentially present in the Cycle 21 core and 8 fuel assemblies currently in the spent fuel pool that are planned for use in the Cycle 22 core. This condition potentially impacts two Updated Final Safety Analysis Report (UFSAR) described SSC design functions. Because of the insufficient anneal, there is a potential for an increased material relaxation rate to result in increased grid-to-rod gaps, which adversely impacts the ability of the grid to perform the function of limiting fuel rod vibration and thereby reduces grid-to-

10 CFR 50.59 Summary of Changes and Evaluations

Parent Document	Change Description	Evaluation Summary
		rod fretting (GTRF) margin. There is also a potential for the ability of the grid to maintain a coolable geometry and a geometry that will ensure control rod insertability to be impacted because of the affect the presence of this material has on grid strength. Although there is a potential impact on grid strength, it was determined to be minimal. Therefore, the grid will still maintain the fuel array in a coolable geometry and preserve control rod insertability. Although there is a potentially minor impact on GTRF, adequate margin exists to offset the impact. It was determined that no changes to the UFSAR are required. Technical Specifications were also reviewed. It was determined that no changes to the Technical Specifications or Bases are required. It is concluded by this 10CFR50.59 evaluation that fuel grids potentially containing insufficiently annealed material can be operated in the Cycle 21 and subsequent cores, and stored in the spent fuel pool without obtaining a License Amendment. Nuclear Design and Analysis has reviewed the Westinghouse evaluations and concurs with their conclusion.