October 15, 2007

Mr. Jeffrey B. Archie
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, ISSUANCE OF REVISED SAFETY EVALUATION REGARDING AMENDMENT NO. 160 (TAC NO. MD3459)

Dear Mr. Archie:

The Nuclear Regulatory Commission (NRC) issued Amendment No. 160 to the Facility Operating License for the Virgil C. Summer Nuclear Station, Unit No. 1 (VCSNS), on August 30, 2002. The amendment and associated Safety Evaluation addressed increasing the capacity of the spent fuel pool storage capacity by replacing the storage racks in the spent fuel pool.

On October 4, 2006, you submitted a letter requesting clarifications to the Safety Evaluation for the purpose of ensuring that the VCSNS licensing basis remains accurate. The NRC staff has reviewed the proposed changes and, where an appropriate basis was provided, has made the change as noted in the enclosed revised Safety Evaluation. The NRC staff reaffirms its overall findings made in the Safety Evaluation that supported the issuance of Amendment No. 160.

Sincerely,

/RA/

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosure: Safety Evaluation

cc w/enclosure: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 160 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated July 24, 2001, as supplemented by letters dated April 4, May 7, June 17, July 2, July 15, and July 25, 2002, South Carolina Electric & Gas Company (SCE&G), the licensee, requested changes to the technical specifications (TS) for the Virgil C. Summer Nuclear Station (VCSNS). The proposed changes would increase the spent fuel pool (SFP) storage capacity by replacing all 11 existing rack modules with 12 new storage racks. The rerack will increase the storage capacity from 1,276 storage cells to 1,712 storage cells. The new racks will have Boral neutron-absorbing material instead of the degrading Boraflex used in the existing racks.

The supplemental letters listed above contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act (Act) requires applicants for nuclear power plant operating licenses to include TS as part of the license. These TS are derived from the plant safety analyses. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.36 contains the Commission's regulatory requirements that are related to the content of the TS. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls.

According to 10 CFR 50.59 (c)(1)(i), the Commission requires a licensee to submit a license amendment pursuant 10 CFR 50.90 if a change to the TS is required. Furthermore, 10 CFR 50.59 requires that U. S. Nuclear Regulatory Commission (NRC) approve the TS changes before the TS changes are implemented. TS changes are not solely justified on the basis of adopting the model Standard TS (STS). In each case the NRC staff makes a determination that the change maintains adequate safety. Changes that result in a relaxation (less restrictive conditions) require detailed justification from the licensee.

SCE&G is revising the TS to reflect the new fuel storage design safety analysis and to continue efficient and safe operation of the plant. The requested changes would also allow credit for soluble boron (for accident cases only) in the SFP criticality analyses. In this submittal, the licensee continues to meet regulatory requirements by performing its criticality analyses of the VCSNS spent fuel storage racks in accordance with 10 CFR 50.68 (b), "Criticality Accident Requirements."

Appendix A of 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," requires that criticality in fuel storage be prevented, preferably by geometrically safe configurations. In addition, 10 CFR 50.68 (b), sets requirements to prevent criticality during fuel handling. Also, other NRC guidance (Reference 2) and the applicable American National Standards Institute (ANSI) standards (Reference 3) establish the criteria for determining the acceptability of the licensee's SFP arrangements. Because of these requirements, the NRC licensed VCSNS with TS 5.6, "Fuel Storage." TS 5.6 required that the spent fuel storage racks consist of 1276 individual cells, grouped into 3 regions. However, VCSNS is projected to lose full-core offload capability in the SFP following Cycle 17, which ends in the spring of 2008. The requested reracking would increase the storage capacity from 1,276 storage cells to 1,712 storage cells, an increase of 436 cells. This additional storage capacity would allow continued full-core offload capability through the end of Cycle 24, in 2018, without any restrictions from SFP storage capacity limitations.

Following the rerack, the licensee will use only two rack types versus the three types currently present in the pool. Region 1 of the SFP will permit storage of 200 assemblies with enrichments up to 4.95 nominal weight percent (w/o) U-235 without regard to fuel burnup. Region 2 will permit storage of 1512 assemblies that meet minimum burnup requirements for unrestricted storage. Due to the increased capacity and boration of the water in the spent fuel storage facility, the licensee is proposing to modify the VCSNS TS to reflect the resulting necessary operational changes. The spent fuel storage redesign resulted in new criteria and graphs for determining fuel burnup times and acceptable fuel assembly locations. Specifically, the licensee will add a new requirement on the TS limit for boron concentration during nonrefueling fuel evolutions.

Additionally, the licensee is moving TS 3.9.10, 3.9.11, and 3.9.12 out of the Refueling Operations section (3.9) of TS into the Plant Systems section (3.7) since they are not specific to refueling operations. This conforms to the Improved Standard Technical Specifications, NUREG-1431 Title. NUREG-1431 uses Section 3.9 for refueling operations and locates fuel handling facility information under Plant Systems in Section 3.7. This modification is a format change to the TS.

3.0 TECHNICAL EVALUATION

3.1 Technical Specifications

3.1.1 Description of Changes - Overview

The licensee modified the VCSNS TS to reflect the operational changes resulting from the increased capacity and the reconfiguration. The spent fuel storage redesign required the development of acceptable fuel assembly location based on burn-up times. These new requirements are reflected in corresponding changes to the VCSNS TS. The licensee revised the current TS to account for the reduction in minimum in-core hold time from 100 hours to 72 hours and to add a new TS requirement on the limit for boron concentration during nonrefueling fuel evolutions. The licensee reevaluated the consequences of a fuel assembly drop in the SPF to incorporate the shorter reactor hold time.

The licensee proposed a new TS that requires a minimum of 500 parts per million (ppm) boron whenever it moves new or irradiated fuel during nonrefueling movements in the SFP, fuel transfer canal, or cask loading pit. This minimum boron concentration will ensure that the fuel remains subcritical under any normal fuel handling or misloading accidents. During refueling operations that involve the movement of fuel in the reactor core, the licensee will maintain a minimum boric acid concentration of 2,000 ppm in the SFP.

3.1.2 Evaluation of Changes

In conducting the review, the staff evaluated each proposed TS change resulting from the design modifications to the spent fuel handling facility. The licensee developed the following proposed TS changes based on the criteria contained in 10 CFR 50.36 while conforming to the format of the Improved Standard Technical Specifications, NUREG-1431. The staff review confirms the acceptability of the changes on those bases. The staff technical evaluation of the proposed changes is provided in Sections 3.2 through 3.8 of this Safety Evaluation.

TS 3/4.9.10, 3/4.9.11, and 3/4.9.12

The licensee proposes to move TS Sections 3/4.9.10, 3/4.9.11, and 3/4.9.12 into the newly created TS Sections 3/4.7.10, 3/4.7.11, and 3/4.7.12, respectively, in order to conform to the format of NUREG-1431. NUREG-1431 uses Section 3.9 for refueling operations and locates fuel handling facility information under Plant Systems in Section 3.7. This is a format or administrative change to the TS that does not change any requirements and is, therefore, acceptable.

TS 3/4.7.12 Spent Fuel Assembly Storage

The licensee proposes to amend the current TS in order to provide revised fuel assembly burnup curves to reflect the change from three region operation to two region operation and other design modifications. The change is acceptable.

TS 3/4.7.13 Spent Fuel Pool Boron Concentration

The licensee added an additional TS to require a minimum of 500 ppm (425 ppm rounded up to 500 ppm) boron whenever new or irradiated fuel is being moved (non-refueling movement) in the SFP, fuel transfer canal, or cask loading pit. The VCSNS current TS do not require a boron concentration limit for non-refueling fuel movement. This 500 ppm minimum boron concentration will ensure that the fuel remains subcritical under any normal fuel handling or misloading/mispositioning accidents. During refueling operations that involve the movement of fuel in the reactor core, a minimum boric acid concentration of 2,000 ppm, or a $K_{eff} \leq 0.95$, will be maintained in the reactor coolant system and refueling canal in accordance with TS 3/4.9.1. Since establishing a TS for boron concentration where there was none is more restrictive, and since the TS value is conservative, the NRC staff finds the request acceptable.

TS 3/4.9.3 Decay Time

The licensee proposes to add a new figure, Figure 3.9-1, for determining the in-core holding time of fuel. The component cooling water system (CCWS) removes decay heat from the reactor core when the reactor is in the shutdown condition. The CCWS water temperature influences the duration needed for the fuel to decay for the safe movement of irradiated fuel. New Figure 3.9-1 shows in-core holding time of fuel based on the CCWS temperature. The new figure resulted from the spent fuel facility design change and replaces the fixed in-core holding time of 100 hours in the current TS.

Additionally, the licensee proposes to reduce the minimum in-core hold time of the fuel from 100 hours to 72 hours. The licensee provided an analysis that satisfactorily demonstrates the adequacy of this minimum in-core hold time and, therefore, the change request is acceptable.

TS 3/4.9.12 Spent Fuel Assembly Storage

The licensee proposes to move the revised Figure 3.9-1 to Figure 3.7-1 because Section 3.7 contains the fuel handling facility information. Additionally, Figure 3.9-2 was deleted since it affects only Region 3 of the spent fuel storage and Region 3 was eliminated by the redesign. These proposed changes are administrative and, therefore, are acceptable.

3.1.3 Conclusion

With the proposed changes to the VCSNS TS, the licensee provides TS that reflect the new fuel handling requirements that resulted from the fuel handling facility redesign. The NRC staff concludes that the proposed changes satisfy 10 CFR 50.36 with regard to the content of TS and conform to the model provided in NUREG-1431. On this basis, the NRC staff concludes that the proposed changes to the VCSNS TS are acceptable.

3.2 Criticality Technical Evaluation

The July 24, 2001, submittal contains the criticality analyses supporting the increase in the SFP storage capacity performed by the Holtec Corporation for VCSNS. The Holtec report contains the criticality analyses accounting for the increase in the storage capacity while maintaining K-effective (K_{eff}) less than or equal to 0.95 under normal and abnormal conditions.

3.2.1 Criticality Calculations Associated with the Rerack Request

The design of the new racks to be installed in the two regions of the SFP will maintain the subcriticality margin when fully loaded with enriched fuel and submerged in unborated water at a temperature corresponding to the highest reactivity. The licensee will use racks incorporating Boral panels instead of Boraflex panels to maintain subcriticality in the SFP.

The criteria that define the maximum permissible reactivity will control the storage of spent fuel in each region. The Region I section of the pool will store the most reactive fresh fuel with an enrichment of up to 4.95 w/o U-235. The Region 2 section of the pool will also be able to accommodate fuel of 4.95 w/o U-235 enrichment, but will be subject to specific burnup limits. If the fuel assembly does not meet the requirements for unrestricted storage in Region 2, it must be stored in Region 1.

The Holtec Corporation analyzed the VCSNS spent fuel storage racks and documented the analyses in Holtec report HI-2012624 (attachment to Reference 1). Even though the NRC has not formally approved the methodology described in the Holtec report, it is generally recognized as the industry standard, and is reviewed and evaluated on a plant-specific basis. The methodology described in the Holtec report ensures that k_{eff} remains less than or equal to 0.95 as recommended in ANSI/American Nuclear Society (ANS)-57-1983 (Reference 3) and NRC guidance (Reference 2). The methodology also takes partial credit for soluble boron in the SFP criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1. K_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95-percent probability, 95-percent confidence (95/95) level as described in the Holtec report; and

2. K_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in the Holtec report.

The licensee performed the analysis of the reactivity effects of fuel storage in the VCSNS SFP using the MNCP4a and KENO-Va codes. The licensee also performed the criticality analysis using the CASMO-4 code. MNCP4a is a continuous energy three-dimensional Monte-Carlo code developed at the Los Alamos National Laboratory. KENO-Va is a three-dimensional multigroup Monte-Carlo code developed at Oak Ridge National Laboratory as part of the SCALE 4.3 package. CASMO-4 is a two-dimensional multigroup transport theory code used for depletion analyses. All these codes are industry standard codes that were validated through benchmarking to relevant critical experiments. The staff has historically found these codes acceptable for licensing applications.

The staff evaluated the licensee's submittal by comparing it to the above methodology and concluded that VCSNS used acceptable codes to perform its calculations. By following the Holtec methodology and by using acceptable codes, the licensee ensured that its proposed changes continue to meet the acceptance criteria for SFP storage. Therefore, the NRC staff finds the licensee's methodology to be acceptable. In addition, this amendment request conforms to General Design Criterion (GDC) 62, which is the basis for the current design.

3.2.2 Summary of Criticality Analysis

The licensee's analysis used the most reactive design and the most reactive temperature to set the storage requirements. The analyses included means to account for the bias and uncertainty associated with the benchmarking of the methodology, a bias for the underprediction of reactivity due to boron particle self-shielding, and the uncertainty due to mechanical tolerances from the manufacturing process. The licensee also included additional uncertainties related to irradiated fuel as described in the burnup credit methodology discussed in the Holtec report. The licensee determined these uncertainties at the 95/95 probability/confidence level, using procedures described in the regulatory guidance of Reference 3. Because the licensee followed the appropriate regulatory procedures and used conservative values for the analyses, the NRC staff finds them acceptable for use.

3.2.2.1 Normal Operating Conditions

The licensee performed criticality analyses for each of the regions. The licensee performed calculations to qualify the Region I racks for the storage of fresh unburned fuel assemblies with the maximum enrichment of 4.95 w/o U-235. The K_{eff} calculated for Region I was found to be less than 0.95, including all the uncertainties and at the 95/95 level. Table 4.2.1 of the Holtec report of Reference 1 documents the criticality analysis for Region I.

The licensee also performed calculations to qualify the Region 2 racks for the storage of spent fuel assemblies with a maximum nominal initial enrichment of 4.95 w/o U-235. These assemblies have an accumulated minimum burnup of 41.6 GWD/MTU or a fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.1 of the submittal. The analyses found the K_{eff} for Region 2 to be less than 0.95, including all the uncertainties and at the 95/95 level. Table 4.2.2 in the Holtec report of Reference 1 documents the criticality analysis and the acceptance criteria for Region 2.

The maximum reactivity associated with those assemblies that qualify for storage in Region 2 includes the effect of axial distribution in burnup and provides an additional reactivity uncertainty for the depletion calculation. Those assemblies that can be stored in Region 2 are subject to the qualification criteria dictated by Figure 4.1.1 and Table 4.2.3 of Reference 1. The NRC staff evaluated the licensee's analysis per the requirements of GDC 62 and of References 2 and 3, and found that the analysis for the normal operating conditions is acceptable since it satisfies the requirement that K_{eff} not exceed 0.95.

3.2.2.2 Abnormal and Accident Conditions

Although the NRC permits credit for the soluble boron (neutron absorber or poison) normally present in the SFP water, most abnormal or accident conditions will meet the limiting reactivity ($K_{eff} \leq 0.95$) even in the absence of soluble poison. The licensee analyzed all postulated accidents (i.e., dropped fuel assembly, water temperature and density effects, eccentric positioning of a fuel assembly within the rack, abnormal and misplacement of fresh fuel assembly, etc.) for this amendment request. The licensee's analysis shows that the abnormal location of a fresh fuel assembly has the potential for exceeding the limiting reactivity (K_{eff} less than or equal to 0.95, but always less than 1.0) should there be a concurrent and independent accident where all the soluble boron had been lost in the SFP. The largest reactivity increase would be caused by an accident where a fresh assembly of the highest permissible enrichment is inadvertently loaded into a Region 2 storage cell with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. Under this accident condition, NRC guidelines permit credit for the presence of soluble boron (Reference 5). Considering these scenarios, calculations by Holtec show that the SFP must have a minimum soluble boron concentration greater than or equal to 400 ppm to maintain a $K_{eff} \leq 0.95$.

In addition, the analysis shows that the misplacement of an assembly outside and adjacent to the Region 2 racks would also lead to a failure in meeting the K_{eff} limit of 0.95. This scenario would occur only if a fresh fuel assembly of the highest permissible enrichment were to be inadvertently placed outside and adjacent to a Region 2 storage cell, with the remainder of the rack fully loaded with fuel of the highest permissible reactivity and no Boral panel between the fuel in the storage rack and the misplaced assembly. Calculations by Holtec show that the SFP must have a minimum soluble boron concentration greater than or equal to 425 ppm to maintain $K_{eff} \leq 0.95$ for this case.

The licensee analyzed its spent fuel storage racks by taking into account boron credit in accordance with the methodology described in the Holtec report (Reference 1). This methodology ensures $K_{eff} \leq 0.95$ as recommended in ANSI/ANS-57-1983 (Reference 3) and NRC guidance (Reference 2). The licensee also analyzed the ability of the SFP storage racks to accommodate assemblies with fuel enrichment up to 4.95 w/o U-235 while maintaining $K_{eff} \leq 0.95$, including uncertainties, tolerances, biases, and credit for soluble boron. The licensee used the soluble boron credit to offset the uncertainties, tolerances, and off-normal conditions and to reduce the K_{eff} to ≤ 0.95. The licensee's analysis showed that the SFP does not require any soluble boron to maintain $K_{eff} \leq 0.95$ under normal conditions and to provide the subcritical condition with a margin of 5 percent, based on a 95/95 probability/confidence level calculation.

The licensee's analyses assumed that the moderator was pure water at a temperature of 68 °F and a density of 1.0 gm/cc. The analyses also included treatment for uncertainties due to tolerances in fuel enrichment and density, storage cell inner diameter, storage cell pitch, stainless steel thickness, assembly position, calculation uncertainty, and axial burnup. The licensee also appropriately determined the uncertainties at the 95/95 probability/confidence level and included a methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of the normal range of SFP water temperatures. The NRC staff evaluated the licensee's analysis per the requirements of GDC 62 and of References 2 and 3. The analysis for the abnormal and accident conditions is acceptable since it satisfies the requirement of $K_{eff} \leq 0.95$ as prescribed per the above regulatory requirements.

3.2.3 Reactivity Equivalence

The concept of reactivity equivalence is predicated upon the reactivity decrease associated with fuel depletion. For burnup credit, a series of reactivity calculations are performed to generate a set of enrichment and fuel assembly discharge burnup ordered pairs that all yield an equivalent K_{eff} when stored in the spent fuel storage racks.

K_{eff} contour plots are generated for all the cell configurations for fuel storage in the high density spent fuel racks. These curves represent combinations of fuel enrichment and discharge burnup that yield the racks' multiplication factor as the racks are loaded with zero burnup fuel assemblies with maximum-allowed enrichments. Uncertainties associated with the burnup credit include a reactivity uncertainty applied linearly to the credit to account for calculation and depletion uncertainties. The NRC staff reviewed the licensee's submittal for reactivity equivalencing uncertainties per the methodology described in NUREG/CR-6683 (Reference 4). Upon review of the licensee's calculations, the NRC staff is satisfied that the licensee included the appropriate uncertainties in all the criticality calculations, and that these criticality calculations were performed in compliance with the methodology described in NUREG/CR-6683. Therefore, the NRC staff finds these calculations acceptable.

3.2.4 Conclusion

The NRC staff finds that the proposed VCSNS license amendment request meets the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The NRC staff concludes that the licensee conducted the necessary analysis in accordance with NRC guidelines and the ANSI standards. The analysis shows that the design and operation of VCSNS will maintain the maximum neutron multiplication factor K_{eff} within the acceptance criteria under all postulated accident conditions.

3.3 Compatibility of Structural Materials and Boral

The new storage racks proposed for use in the SFP are manufactured by Holtec International. These freestanding, self-supporting racks are designed to stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*.

3.3.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include ASME SA240-304L for all sheet metal stock and internally threaded support legs, ASME SA564-630 precipitation-hardened stainless steel (heat treated to 1100 °F) for the externally threaded support spindle, and ASME Type 308L for weld material. These materials used in the Holtec racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the spent fuel environment. Therefore, they are acceptable for use in this application.

3.3.2 Poison Material

The Holtec racks employ Boral as the neutron-absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in SFP environments where it has maintained its neutron-attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels, resulting in deformation of the storage cells. The racks are designed to vent the corrosion gases to prevent the deformation of the storage cells. The neutron-absorbing capability of Boral is not affected by this corrosion process.

3.3.3 Conclusion

Based on its evaluation, the NRC staff finds that the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec International are compatible with the SFP environment at VCSNS. The passivation exhibited by the racks does not affect their neutron-absorbing capability. The NRC staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

3.4 Structural Integrity

This section describes the staff's review to assure the structural integrity and functionality of the racks and the stored fuel assemblies subject to the effects of the postulated loads discussed in Appendix D of Standard Review Plan (SRP) Section 3.8.4, including Fuel Handling Accident (FHA) loads.

3.4.1 Storage Racks

The spent fuel racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). SCE&G, with its contractor Holtec, performed structural analyses of the racks.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the VCSNS spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are freestanding and self-supporting equipment, and are not attached to the floor or walls of the SFP structure. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements, and friction elements, as defined in the DYNARACK program, was used to simulate the three-dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies, including frictional and hydrodynamic effects. The program was utilized to calculate nodal forces and displacements at the nodes, and to obtain the detailed stress field in the rack elements from the calculated nodal forces.

Analyses of two models were performed: a 3-D single rack (SR) model and a 3-D multirack (MR) model. For the 3-D MR analyses, all racks were considered to be fully loaded with four different coefficients of friction (μ=0.2, 0.5, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the SFP floor. The 3-D MR analyses were performed to investigate the fluid-structure interaction effects between the racks and the SFP walls as well as those among the racks and to identify the worst-case response for rack movement and for rack member stresses. For the 3-D SR analyses, the rack was considered to be fully loaded, half loaded and almost empty with a coefficient of friction (μ= a random value where the mean is about 0.5) between the rack pedestal and the SFP floor. The 3-D SR analyses were performed to investigate the stability of the rack with respect to overturning.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal-acceleration components and one vertical-acceleration component) were generated from the design response spectra defined in the Final Safety Analysis Report (FSAR Reference 6). SCE&G demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra and matching a target power spectral density function compatible with the design response spectra as discussed in SRP Section 3.7.1.

A total of 165 3-D SR and MR analyses were performed. The racks were subjected to the service, upset, and faulted loading conditions (Level A, B, and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 1.52 inches, indicating that there is an adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the rack and the SFP wall. However, the results show that there is impact potential between the racks. The NRC staff compared the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension with corresponding allowable stresses specified in the 1989 Edition of ASME *Boiler and Pressure Vessel Code*, Section III, Subsection NF. The stress results show that the induced impact forces under the SSE loading condition are small, and all induced stresses in the racks are smaller than the corresponding allowable stresses specified in the ASME *Boiler and Pressure Vessel Code,* indicating that the rack design is adequate.

SCE&G also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions. SCE&G demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, thus indicating that the weld connection design of the rack is adequate.

Based on (1) SCE&G's parametric evaluation (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when these stresses are compared to the corresponding allowables provided in the ASME *Boiler and Pressure Vessel Code*, and (3) SCE&G's overall structural integrity conclusions supported by both SR and MR analyses, the NRC staff concludes that the rack modules will perform their safety function of maintaining their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.4.2 Spent Fuel Storage Pool

SCE&G analyzed the SFP structure to demonstrate the adequacy of the structure under fully loaded fuel racks with all storage locations occupied by fuel assemblies. The fully loaded structure was subjected to the load combinations specified in the VCSNS FSAR (Reference 2).

Reference 1 indicates that the induced stresses due to the racks in the SFP structure are smaller than corresponding allowable stresses of the American Concrete Institute 349, “Code Requirements for Nuclear Safety-Related Concrete Structures.” In view of SCE&G’s stress calculations, the NRC staff concludes that SCE&G’s structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading, and SSE loading conditions. Thus, the SFP structural design is acceptable.

3.4.3 Fuel Handling Accident

The following three refueling accident cases were evaluated by SCE&G: (1) drop of a fuel assembly with its handling tool that impacts the baseplate (deep drop scenario); (2) drop of a fuel assembly with its handling tool that impacts the top of a rack (shallow drop scenario); and (3) drop of a rack from the height of 45 feet that impacts the liner plate.

The analysis results for the first accident case show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads; therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results for the second accident drop case show that damage will be restricted to a depth of 2.55 inches below the top of the rack, thus indicating that the evaluation satisfies the acceptance criteria presented in the criticality safety evaluation (Reference 1). The analysis results for the third accident case show that the liner plate is locally damaged.

However, the analysis for the integrity of the pool slab indicates that a primary failure in the SFP will not occur. The NRC staff reviewed SCE&G's analysis results in Reference 1 and agrees with its findings. This evaluation is acceptable based on SCE&G's structural integrity conclusions supported by the parametric evaluations.

3.4.4 Conclusion

Based on its review of SCE&G's submittal (Reference 1), the NRC staff concludes that SCE&G's structural analysis and the structural design of the spent fuel rack modules and the SFP structure are adequate to withstand the effects of the applicable loads, including the SSE effects. The analyses and design are in compliance with the current licensing basis set forth in the FSAR (Reference 2) and applicable provisions of the SRP and are, therefore, acceptable.

3.5 Radiation Protection

3.5.1 Occupational Radiation Exposure

The NRC staff has reviewed the licensee's plan for the modification of the VCSNS spent fuel racks with respect to occupational radiation exposure. Based on the lessons learned from a number of facilities that have performed similar operations in the past and their experience with reracks, the licensee estimates that the collective occupational worker dose for the proposed fuel rerack project will be between 6 and 12 person-rem.

All of the operations involved in the removal of existing racks and the installation of the new fuel racks will be governed by procedures. These procedures are based on the principle of keeping doses as low as reasonably achievable (ALARA), consistent with the requirements of 10 CFR Part 20. The radiation protection department will prepare a radiation work permit (RWP) for the various in-pool and out-of-pool jobs. The RWP and supporting job procedures will establish requirements for timely external radiation and airborne surveys, personal protective clothing and equipment, individual monitoring devices, and other access and work controls consistent with good radiation protection practices and 10 CFR Part 20 requirements. Continuous health physics technician (HPT) coverage will be provided and maintained when a diver is in the pool, and when any potentially contaminated object is being removed from the pool. Each member of the project team will receive radiation protection training on the reracking operations, consistent with the requirements of 10 CFR Part 19. Project-specific training will include hot particle hazards and the potential for extremity doses from working in the fuel pool or with the old racks (e.g., decontaminating and packaging them for shipment off-site). Prior to the start of the job, lessons learned from previous pool rerackings will be discussed as part of the ALARA briefing. Daily pre-job briefings, which will include information on pertinent ALARA issues, will be used to inform workers and HPTs of job scope and techniques. All divers will be fully trained and qualified for nuclear diving.

For out-of-pool work activities, all workers will be provided with thermoluminescence dosimeters (TLDs) and electronic alarm dosimeters. Additional personal monitoring devices (e.g., extremity badges) will be used, as appropriate. Periodic radiation surveys will be conducted for direct radiation levels and loose surface contamination levels, as appropriate and in accordance with the governing RWP. Historical experience during similar reracking shows that radioactive airborne material levels in the above-pool work area should be negligible during the rerack job. However, air sampling will be performed and continuous air monitors will be used when a job evolution has the potential for generating significant airborne radioactivity. Personal respiratory equipment will be available, if needed. In order to minimize contamination and airborne problems, all equipment removed from the pool will be surveyed before removal, surveyed as it breaks the water surface, rinsed off, and resurveyed by or under the direction of a qualified HPT.

The VCSNS SFP rerack project will use qualified divers for both rack removal and installation. No divers will be allowed in the SFP during any movement of spent fuel to ensure that these divers are not exposed to high and very high radiation sources (e.g., spent fuel). All diving operations will be governed by special procedures that will require extensive surveys of the dive area before dives, and the divers will be trained to use calibrated underwater radiation survey instruments for confirmatory surveys of their work area. The location of significant radiation sources will be made known, to the divers, and the divers' range of motion in the SFP will be restricted by a tether, that will help ensure that a diver does not get too close to high and very high radiation sources. Additionally, underwater barriers will be used to physically define the safe dive area. No deviations from the planned, prescribed dive will be allowed. Continuous audio and video monitoring and communication will be in place to allow for constant poolside surveillance of all diver activities. If any of these monitoring capabilities are lost, the dive will be terminated. Due to the steep dose gradients from water shielding, each diver will be provided with multiple TLDs and electronic dosimeters for whole body and extremity monitoring, with continuous remote dose rate readouts for poolside observation, monitoring, and control. The VCSNS diving control and survey procedures described above meet the intent of Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," Appendix A, "Procedures for Diving Operations in High and Very High Radiation Areas." This appendix was developed from the lessons learned from previous diver overexposures and mishaps, and summarizes good operating practices for divers that are acceptable to the NRC staff.

An underwater vacuum system will be used to supplement the installed SFP filtration system so that the levels of radiation and contamination including hot particles and debris can be reduced before diving operations. The SFP floor dive area will be vacuum-cleaned with long-handled tools from above the pool. Final radiation surveys and visual inspection by underwater camera will be performed before any diving activities. These actions to identify and control hot particles and debris should effectively minimize the potential for unplanned diver exposures from these sources.

Before the old fuel racks are removed from the pool, they will be cleaned underwater using high-pressure washing. After cleaning, while the racks are still over the pool, radiation surveys will be performed to determine if further decontamination is needed before the racks are prepared for shipment off-site. The racks will be bagged remotely to minimize potential worker contamination and maintain doses ALARA. Once properly packaged in approved shipping containers, the racks will be shipped in accordance with Department of Transportation and NRC regulations. The licensee will use the existing SFP filtration system during fuel rack installation to maintain water clarity in the SFP. These engineering controls and handling procedures will help minimize the spread of contamination (e.g., hot particles), while keeping worker doses ALARA.

The storage of additional spent fuel assemblies in the SFP and the reduction in minimum cooling time from 100 hours down to 72 hours before fuel movement will result in negligible increases in the external dose rates on the refueling floor and in accessible areas adjacent to the SFP. Existing normally accessible areas around the fuel storage pool are designated Radiation Zone II. That designation will be maintained with the external dose rates remaining less than 2.5 mrem/hr. The maximum dose rates outside the concrete walls of the SFP will remain less than 0.01 mrem/hr. The area most impacted by the pool rerack is the fuel transfer canal (FTC), assuming it to be drained and empty. Assuming an empty FTC, to keep radiation levels below 2.5 mrem/hr, procedures will require that no fuel except old fuel be stored in the first five rows closest to the FTC gate slot. Normally, the FTC will be filled with water.

On the basis of its review of the VCSNS proposal, the NRC staff concludes that the SFP rerack can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The NRC staff finds the projected dose for the project of about 6 to 12 person-rem to be appropriate and in the range of doses for similar SFP modifications at other plants and, therefore, acceptable.

3.5.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP purification system. The licensee predicts that the installation of the new racks will generate slightly more resin from the new, increased capacity rack installation; therefore, the licensee may more frequently change-out the SFP purification system during the reracking operation. In order to keep the SFP water reasonably clear and clean and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP as necessary to remove any radioactive crud, sediment and other debris before the new fuel rack modules are installed. The filters from this underwater vacuum will be a minor source of solid radioactive waste. However, the licensee does not expect that the increase in storage capacity of the SFP will result in a significant change in the long-term generation of solid radioactive waste at VCSNS.

The disposal of the used spent fuel racks will result in a one-time incremental increase in solid waste. Because ongoing volume reduction efforts have effectively minimized the amount of waste generated, this incremental 1-year increase is bounded by the plant's original licensing basis described in the Final Environmental Statement and, therefore, is acceptable.

3.5.3 Gaseous Radioactive Wastes

The storage of additional spent fuel assemblies in the SFP is not expected to affect the releases of radioactive gases from the SFP. Gaseous fission products such as krypton-85 and iodine-131 are produced by the fuel in the core during reactor operation. Small amounts of these fission gases are released to the reactor coolant from the small number of fuel assemblies that develop leaks during reactor operation. During refueling operations, some of these fission products enter the SFP and are subsequently released into the air. There will be not be an increase in the amounts of gaseous fission products released to the atmosphere as a result of the increased SFP fuel storage capacity becuase the frequency of refuelings and the number of freshly off-loaded spent fuel assemblies stored in the SFP, at any one time, will not increase.

The increased heat load on the SFP from the storage of additional spent fuel assemblies could potentially increase the SFP evaporation rate. However, based on previous reracks at other facilities, this increased evaporation rate is not expected to significantly increase, the amount of gaseous tritium released from the pool. Thus, the licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to significantly increase due to the expanded SFP storage capacity. This is consistent with the operating experience with previous SFP expansions to date. Gaseous effluents from the spent fuel storage area are combined with other station exhausts and monitored before release. Past SFP area contributions to the overall site gaseous releases have been insignificant and should remain negligible with the increased capacity. The impact of any increases in site gaseous releases should be negligible, and the resultant doses to the public will remain very small fractions of the 10 CFR Part 20 and 10 CFR Part 50, Appendix I, dose limits.

3.5.4 Liquid Radioactive Wastes

The release of radioactive liquids will not be affected directly as a result of the SFP expansion. The SFP ion exchanger resins remove soluble radioactive materials from the SFP water. When the resins are changed out, the small amount of resin sluice water is processed by the radioactive waste system before release to the environment. As stated above, the frequency of resin changeout may increase slightly during the installation of the new racks. However, the amount of liquid effluents released to the environment as a result of the proposed SFP expansion is expected to be negligible.

3.5.5 Radiological Impact Assessment

Radiation protection personnel will monitor the doses to the workers during the SFP reracking operation, and all work will be in accordance with RWPs and implementing procedures. Divers will be used for the SFP racking operations, and the licensee will provide procedures specifying required survey, personal dosimetry, and other work requirements and controls that meet the intent of Regulatory Guide 8.38, Appendix A guidance. The total occupational dose to plant workers as a result of the SFP expansion operation is estimated to be between 6 and 12 person-rem. This dose estimate is reasonable, given the work scope proposed, and is consistent with comparable doses for similar SFP projects performed at other plants. The SFP rack project will follow detailed procedures prepared with full consideration of ALARA principles, consistent with the requirements of 10 CFR Part 20.

On the basis of our review of the licensee’s proposal, the NRC staff concludes that the VCSNS SFP rerack can be performed in a manner that will ensure that doses to workers will be maintained ALARA. The estimated collective dose to perform the proposed SFP racking operation is a small fraction of the annual collective dose accrued at the facility.

3.6 Radiological Consequence Analyses

The NRC staff reviewed the changes proposed by SCE&G in its submittal of July 24, 2001, with additional information submitted by letters dated May 7 and July 15, 2002. The NRC staff did confirmatory calculations for the design-basis FHA. The licensee stated, and the NRC staff concurs, that the FHA is the limiting event with regard to the proposed TS changes. Table 1 tabulates the analysis inputs and assumptions found acceptable to the NRC staff. Although the NRC staff performed confirmatory analyses, the staff’s approval of this amendment is based on the information docketed by the licensee and on the NRC staff’s finding that the methods, inputs, and assumptions used in the licensee’s analyses are acceptable.

3.6.1 FHA Radiological Consequences

The design-basis FHA analysis postulates that a spent fuel assembly is dropped during refueling, damaging all of the rods in the assembly plus 50 additional rods in an adjacent assembly (a total of 314 rods). The accident analysis assesses whether design features for mitigating environmental releases meet certain design criteria. At VCSNS, this accident could happen inside the containment (CNMT) or in the fuel handling building (FHB), and SCE&G has evaluated both cases.

The SCE&G analyses assume that core inventory is based on 5 w/o initial enrichment fuel and extended operation at 2958 MWt power. The core inventory was determined using the NRC-sponsored SCALE computer code suite. SCE&G considered five fuel burnup exposures ranging from 35,000 MWt/MTU to 70,000 MWt/MTU. (The staff does not address operation above a burnup of 62,000 MWt/MTU.) Since individual radionuclides reach peak equilibrium values at different rates, the highest specific inventory of each contributing radionuclide in any of the burnup ranges was used in the analyses. A decay period of 72 hours between reactor shutdown and fuel movement was assumed. Since the power level and, hence, the inventory in each assembly varies across the core, a radial peaking factor of 1.7 is applied to the average core inventory. SCE&G assumed that 12 percent of the I-131 inventory of the core was in the fuel rod gap, along with 30 percent of the Kr-85, and 10 percent of all other iodines and noble gases. The radioiodine in the gap was assumed to be 99.75 percent elemental and 0.25 percent organic forms.

SCE&G assumes that all of the gap inventory in the 314 damaged fuel rods is instantaneously released through the water in the reactor cavity or SFP into the CNMT or FHB, respectively. SCE&G assumes that 100 percent of the activity release to the CNMT or FHB is released to the environment in 2 hours. Credit was taken for the FHB purge exhaust charcoal filters, but no credit was taken for the reactor building purge exhaust charcoal filters.

Details on the assumptions found acceptable to the NRC staff are presented in Table 1. The offsite doses estimated by the licensee for the postulated FHAs were found to be acceptable.

3.6.2 Control Room Habitability

SCE&G evaluated the dose to operators in the control room. For both cases, the licensee assumed that no automatic isolation of the control room would occur. Instead, an operator-initiated manual actuation was assumed at 10 minutes for the CNMT case and 60 minutes for the FHB case. There is provision for an automatic actuation caused by a high radiation alarm on monitor RM-A1. However, this actuation does not meet engineered safeguards feature requirements for redundancy, and it is appropriate that credit not be taken for automatic actuation. The isolation can be initiated within the control room with a small number of operator actions. During refueling operations, continuous communications are maintained between the control room and the refueling crew. During an FHA only limited actions are required of the control room operators. Therefore, the NRC staff finds the assumed isolation delays acceptable.

Prior to isolation, outside air would be drawn into the control room at a flow rate of 1000 cfm. Upon isolation (at 10 minutes or 60 minutes, as appropriate), the normal makeup airflow is terminated, and a filtered pressurization flow of 1000 cfm commences. Also, the control room air would be recirculated through charcoal filters at a flow rate of 18,143 cfm. The charcoal efficiency is taken as 95 percent for all species of iodine. With the pressurization flow operational, the licensee assumed an unfiltered inleakage flow rate of 10 cfm associated with normal ingress and egress. The NRC staff requested that SCE&G provide additional information to substantiate this inleakage assumption. In its response to this request, SCE&G described its assumption as the current licensing basis value for VCSNS and noted that the NRC staff in two licensing actions prior to 1994 had accepted this value. Integrated testing at several U.S. power reactors has shown leakage exceeding that assumed in control room habitability analyses. Therefore, the control room envelope (CRE) licensing and design bases and applicable regulatory requirements may not be met. The testing experience also indicated that the typical pressurization surveillance test might not be reliable in identifying sources of unfiltered inleakage. The NRC has conducted several public meetings with its stakeholders on this issue since 1998 and recently published a series of proposed draft regulatory guides and a proposed generic communication on control room habitability (67 FR 31385). The intent of the final generic communication will be to formally alert licensees of the NRC staff's findings related to inleakage testing and to request licensees to submit information that demonstrates that the facility CRE complies with current licensing and design basis and applicable regulatory requirements.

SCE&G has not performed an integrated test of the VCSNS CRE to confirm that no unrecognized inleakage paths exist. In its July 15, 2002, letter, SCE&G describes its CRE configuration with regard to inleakage paths. SCE&G notes that it is anticipating the issuance of the forthcoming generic communication while preparing for the performance of an integrated tracer gas test and, if necessary, will update affected radiological and toxic chemical analyses. SCE&G also has an FSAR commitment to maintain a supply of potassium iodine pills for use as a thyroid prophylaxis. The control room staff has immediate access to this supply. SCE&G has stated that sensitivity analyses show that significant increases in the assumed inleakage value can be tolerated without exceeding the GDC 19 acceptance criteria for control room doses. The NRC staff considered this information and has concluded that there is adequate assurance that the radiation doses to the control room personnel will not impede response actions necessary to protect the public. The NRC staff's acceptance of the licensee's unfiltered inleakage assumption is limited to this licensing action and does not exempt the licensee from future regulatory actions that may become applicable due to the generic communication.

Details on the assumptions found acceptable to the NRC staff are presented in Table 1. The doses estimated by the licensee for the postulated FHA were found to be acceptable.

3.6.3 Atmospheric Relative Concentration Estimates

The FHA dose analyses results described in the May 7, 2002, letter were obtained using control room X/Q values that differed from those in the current approved licensing basis. The X/Q values for the FHA inside CNMT were determined using a puff release model described in a draft regulatory guide (DG-1111) issued for public comment in December 2001. The X/Q values for the FHA outside CNMT were determined using the NRC-sponsored ARCON96 model. Guidance on the use of the ARCON96 model is provided in draft regulatory guide DG-1111. SCE&G did not identify this proposed change of analysis methodology in its July 24, 2001, submittal.

Although the staff expects that a puff model will be included in the final version of the regulatory guide, public comments have indicated a need for some changes in the guidance. The NRC staff determined that it would not be possible to complete its review of the proposed SCE&G model in the timeframe allowed by the licensee's requested completion date for this licensee amendment request. The staff has approved the use of ARCON96 for several licensees. However, additional information would be needed to complete this review. In a telephone conversation on July 2, 2002, the NRC staff notified SCE&G of this situation and asked that SCE&G reconsider the use of the new methodologies. In a telephone conversation on July 9, 2002, SCE&G notified the NRC staff of its intent to modify its May 7, 2002, response to reflect reanalyses using the Murphy-Campe methodology that is the current licensing basis. SCE&G provided the updated results and assumptions using its current licensing basis methodology in a followup letter dated July 15, 2002. The staff used the results documented in the July 15, 2002, submitted to develop its findings.

3.6.4 Technical Specification Changes

3.6.4.1 The licensee proposed revising TS 3/4.9.12, "Spent Fuel Assembly Storage," to update Figure 3.9-1, "Required Fuel Assembly Burnup as a Function of Initial Enrichment to Permit Storage in Region 2," and to delete Figure 3.9-2, "Required Fuel Assembly Burnup as a Function of Initial Enrichment to Permit Storage in Region 3." This TS will be renumbered as TS 3/4.7.12.

This change reflects the SFP configuration following expansion. There is no Region 3 in the new arrangement. There are no impacts on the previously analyzed doses due to this change and, as such, the proposed change is acceptable from an accident radiological consequence perspective.

3.6.4.2 The licensee proposed a new TS 3/4.7.13, "Spent Fuel Pool Boron Concentration," to establish an LCO for the boron concentration assumed in the criticality analyses.

There are no impacts on the previously analyzed doses due to this change since criticality is not assumed and, therefore, the proposed change is acceptable from an accident radiological consequence perspective.

3.6.4.3 The licensee proposed revising TS 3/4.9.3, "Decay Time," to reduce the minimum decay time from 100 hours to 72 hours and to add a new graph of in-core hold time vs. CCWS temperature.

The analysis submitted as part of this license amendment request assumed a 72-hour decay period. Therefore, fuel movement occurring after the proposed 72-hour decay period has been analyzed. The proposed change is acceptable from a radiological standpoint since it is consistent with the analysis assumptions used in demonstrating compliance with radiological acceptance criteria.

3.6.4.4 The licensee proposed renumbering TS 3/4.9.10 as 3/4.7.10, TS 3/4.9.11 as 3/4.7.11, and TS 3/4.9.12 as 3/4.7.12. The licensee also proposed conforming changes to the TS Bases.

These changes are editorial in nature and cannot impact the previously analyzed doses and, as such, the proposed changes are acceptable from an accident radiological consequence perspective.

3.6.5 Conclusion

The NRC staff has reviewed the radiological consequences of the SFP expansion proposed by SCE&G for VCSNS. The NRC staff also reviewed the proposed changes to the TS associated with this license amendment request. In doing this review, the NRC staff relied on information placed on the docket by the licensee, on staff experience in doing similar reviews, and where deemed necessary, on staff confirmatory calculations. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes, and finds that the licensee used acceptable analysis methods and assumptions. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations and finds, with reasonable assurance, that the licensee's estimates of the radiation doses due to the postulated FHA will comply with the requirements of 10 CFR 100.11 and 10 CFR Part 50 Appendix A, GDC 19.

3.7 Heavy Loads

By letter dated January 25, 1985, SCE&G provided their final response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This letter established the current licensing basis for control of heavy loads in the spent fuel storage area at VCSNS. With the exception of handling the fuel transfer canal gates, handling of heavy loads within 15 feet of the SFP is outside of the station's current licensing basis. Therefore, the guidance contained in NUREG-0612 is applicable to evaluation of the temporary gantry crane and the proposed rack installation and removal activities.

3.7.1 Control of Heavy Loads

General and specific SFP area guidance that provides defense-in-depth against heavy load handling accidents is contained in Sections 5.1.1 and 5.1.2 of NUREG-0612, respectively. Section 5.1.1 general guidelines include the following: (1) development of safe load paths; (2) development of procedures for load handling operations; (3) training and qualification of crane operators in accordance with ANSI/ASME B30.2-1976, "Overhead and Gantry Cranes;" (4) design and use of special lifting devices in accordance with ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Container Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials;" (5) selection and use of other lifting devices in accordance with ANSI B30.9-1971, "Slings;" (6) inspection, testing, and maintenance of cranes in accordance with ANSI/ASME B30.2-1976; and (7) design of cranes in accordance with ANSI/ASME B30.2-1976 and CMAA-70, "Specifications of Electric Overhead Travelling Cranes." Section 5.1.2 guidelines provide defense-in-depth for heavy load handling in the SFP area by employing one or more of the following measures: (1) further reducing the probability of a load drop through crane design enhancements, (2) reducing the probability of a load drop affecting a critical component through interlocks or operational controls, and (3) evaluating the consequences of potential load drops to ensure an adequate margin of safety is maintained.

In Attachment IV to its letter dated July 24, 2001, SCE&G stated that the general guidelines of Section 5.1.1 of NUREG-0612 would be satisfied as follows: (1) safe load paths will be defined for movement of the fuel storage racks; (2) all phases of rack installation activities will be conducted in accordance with approved procedures under supervision of a designated individual; (3) all crew members involved in the use of lifting and upending equipment will be trained and qualified using a program that satisfies the guidelines of ANSI/ASME B30.2-1976; (4) special lifting devices employed in the rack lifts will meet the guidelines of ANSI N14.6-1993; (5) other lifting devices will be selected, inspected, and maintained in accordance with ANSI B30.9 - 1971; (6) cranes will be inspected, tested, and maintained in accordance with ANSI/ASME B30.2-1976, with a minor reduction in the scope of testing as described in their letter dated July 25, 2002; and (7) the design of the temporary crane will meet the guidelines of ANSI/ASME B30.2-1976 and CMAA-70. This approach fully satisfies the criteria of Section 5.1.1 of NUREG-0612 and is acceptable.

By letters dated July 2, 2002, and July 25, 2002, SCE&G provided additional information regarding measures that provide defense-in-depth for heavy load handling in and around the SFP. These measures ensure that the structural integrity of the crane will be maintained during heavy load movement. Heavy loads will not be lifted over fuel. The maximum practicable separation between heavy loads and stored fuel will be maintained, and the integrity of the SFP structure will be maintained in the unlikely event of a heavy load drop.

As described previously, a temporary gantry crane will be used for all heavy load lifts in the SFP area because the existing FHB crane cannot reach over the SFP. The temporary gantry crane will travel on the rails for the existing fuel handling bridge crane and the crane trolley will travel the entire width of the SFP. Although heavy loads are not planned to be lifted directly over stored fuel, the non-safety-related gantry structure will travel over safety-related equipment and stored fuel. In addition to the general guidelines of Section 5.1.1 of NUREG-0612, the following measures will be employed to ensure the crane structure will retain its integrity under normal and accident conditions: (1) the crane will be fabricated under the same quality assurance requirements applied to fabrication of safety-related components by the contractor (Holtec); (2) the crane will be designed and analyzed to satisfy the acceptance criteria of the ASME Boiler and *Pressure Vessel Code*, Section III, Subsection NF, “Supports,” for loading combinations including the operating basis and design-basis earthquake accelerations; and (3) the crane will be designed with a wide base and will be operated under administrative controls to ensure a load hangup does not topple or overstress the crane structure.

To ensure heavy loads will not be lifted over fuel, SCE&G states that the racks will be moved at a minimum height above the pool floor along predefined safe load paths to a predefined lift location. Stored fuel will be shuffled into planned configurations to maintain the maximum practicable separation (at least 3 feet) between heavy loads and stored fuel. New racks will be immediately lowered to a minimal height above the pool floor once the rack clears the pool perimeter and any pool wall protrusions. These measures ensure that the potential for damage to fuel will be maintained at an extremely low level throughout the rack replacement evolution.

Finally, SCE&G established by analysis that the integrity of the SFP structure will be maintained in the unlikely event of a heavy load drop. The evaluation concluded that a vertically oriented rack dropped from above the pool would not damage the pool structure. The design of the rack with vertical cooling channels ensures that the rack would strike the pool bottom in a vertical orientation. Impact with a nearby rack or the pool wall would reduce the energy of the impact with the pool bottom, so a direct vertical drop to the pool floor bounds the effects of other potential load drops. Although the load drop could damage the steel pool liner, normally closed valves would limit the total leakage from the pool, and procedures and permanently installed instrumentation are available to ensure operators initiate approriate corrective actions. Therefore, potential damage to the SFP from an accidental load drop would be extremely unlikely to uncover the stored fuel.

Based on the above evaluation, SCE&G has proposed the use of equipment and operational controls such that the potential for a heavy load drop that would strike sensitive components (i.e., stored fuel or sections of storage racks containing stored fuel) will be extremely small. Also, the licensee has provided an acceptable evaluation demonstrating that potential damage to the SFP from an accidental load drop would be extremely unlikely to uncover the stored fuel. Therefore, the storage rack replacement to support revision to TS 5.6.3 is acceptable.

3.7.2 Conclusion

Based on the NRC staff's review and evaluation of the licensee's submittal, the staff concludes that, pursuant to 10 CFR 50.92, the analyses of the spent fuel rerack modules and SFP are in accordance with current industry practice and are consistent with the acceptance criteria set forth in the FSAR. The proposed revisions to TS 3/4.9.3 and TS 5.6.3 are acceptable. The licensee has proposed the use of appropriate equipment and operational controls to safely remove the existing storage racks and install the new storage racks consistent with the intent of guidelines in Section 5.1 of NUREG-0612.

3.8 Decay Heat

The proposed amendment includes two changes that increase the maximum decay heat generation within the SFP. The increase in spent fuel storage capacity results in a small increase in maximum potential decay heat generation due to an increase in the number of older fuel assemblies that can be stored in the pool. Concurrent with the storage expansion, SCE&G proposes to reduce the required minimum decay time prior to fuel movement from 100 hours to a time period dependent on base heat removal capacity but not less than 72 hours. The reduction in minimum decay time creates the potential for a significant increase in decay heat generation within the pool.

Decay heat is removed from the SFP by the spent fuel cooling system. The spent fuel cooling system consists of two safety-related cooling trains, each of which has one pump and one heat exchanger in each train. Heat is removed from the spent fuel cooling system heat exchangers by the safety-related component cooling water system (CCWS). Section 3.1.2 of the VCSNS FSAR describes the extent to which structures, systems, and components important to safety satisfy the GDC contained in Appendix A to 10 CFR Part 50. GDC-61 specifies, in part, that fuel storage systems shall be designed with:

> a residual heat removal system capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and
>
> the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

Section 3.1.2 of the FSAR describes how the spent fuel cooling system provides cooling to remove residual heat from the fuel stored in the SFP and how the system is designed with redundancy to assure continued heat removal. The FSAR also describes that the SFP is designed such that no postulated accident could cause an excessive loss of coolant inventory. These descriptions provide requirements applicable to evaluation of proposed increases in decay heat generation within the SFP.

3.8.1 Spent Fuel Pool Cooling

The proposed change to TS 3.9.3, "Decay Time," that reduces the minimum decay time from 100 hours to a value that allows for adequate heat removal capacity, but not less than 72 hours, and the proposed change to TS 5.6.3, "Fuel Storage - Capacity," that increases the SFP storage capacity limit from 1276 to 1712 fuel assemblies, will result in the potential for increased decay heat loads within the SFP. The availability of adequate heat removal capability was determined by analysis based on the decay time necessary to maintain the SFP temperature below 170 degrees Fahrenheit (°F) at various component cooling water temperatures. This pool temperature was selected as an acceptable value based on evaluation of cooling system performance and the pool structural evaluation. The analyses used the following assumptions:

(1) a full-core offload begins at the minimum decay time and is completed 20 hours later.

(2) the new pool storage racks are filled with previous discharges from 18-month operating cycles; the assumed total fuel inventory conservatively exceeds the proposed limit of 1712 assemblies.

(3) the decay heat is calculated using the ORIGEN2 code assuming a 2-percent thermal power uncertainty and using the licensed thermal power at the time of discharge for historical discharges.

(4) a single SFP cooling train is in service at its maximum flow rate of 2400 gallons per minute (gpm) with heat exchanger fouling and tube plugging at their design values.

At a component cooling water temperature of 89.4 °F or lower, the minimum decay time is not limited by the heat removal capability of the spent fuel cooling system. Instead, the decay time is set by the 72-hour decay time assumed for the FHA analysis. Above a component cooling water temperature of 89.4 °F, the calculated minimum decay time increases with temperature up to 146 hours at the design maximum component cooling water temperature of 105 °F.

In addition to the above analyses, SCE&G provided the results of SFP maximum temperature evaluations for cases involving a partial-core offload, a planned full-core offload, and an unplanned full-core offload 36 days after a planned refueling offload. These evaluations used the same assumptions as the above analyses, with the exception that the minimum decay time was assumed to be 72 hours, the component cooling water temperature was assumed to be 105 °F, and the SFP cooling system is assumed to be operating at the design flow rate of 1800 gpm. In addition, both trains of SFP cooling were assumed to be operating during the full-core offloads. These evaluations resulted in maximum predicted SFP temperatures of approximately 150 °F for all three cases. These results are conservative because they are based on the minimum decay time combined with the maximum component cooling water temperature, a condition that would be precluded in practice by the proposed TS 3.9.3.

The staff performed independent calculations of decay heat load and heat exchanger performance to verify the accuracy of the analyses provided by SCE&G. The decay heat load calculations used the method described in Branch Technical Position ASB 9-2 from NUREG-0800, "US NRC Standard Review Plan," and the heat exchanger performance evaluation used the temperature effectiveness method and heat exchanger performance data from the VCSNS Updated FSAR. These independent calculations, with consideration for the differing analytical methods and assumptions, confirmed the results provided by SCE&G were accurate.

The Updated FSAR states that the typical refueling practice at VCSNS is to perform a full-core offload with two spent fuel cooling pumps in operation. In its letter dated July 2, 2002, SCE&G confirmed this practice and stated that plant procedures require that both spent fuel cooling loops be available prior to the start of the core offload. If indicated pool temperature exceeds 120 °F, and a spent fuel cooling pump becomes unavailable, plant procedures require that flow in the operating loop be increased to 2400 gpm. Under these operating conditions, the staff calculations indicate the SFP temperature would be below 140 °F with both cooling loops in operation. As evaluated above, if one cooling loop failed, SFP temperature would increase but remain below 170 °F. For the condition where only one cooling loop is in operation, SCE&G provided the results of a computational fluid dynamic model of the SFP rack that indicated sufficient natural circulation flow would develop through the cell with the highest rate of heat generation to maintain the coolant subcooled.

Because the spent fuel cooling system consists of two safety-related, independent trains, failure of more than one cooling loop is unlikely. Nevertheless, the available makeup capacity from the refueling or reactor makeup water storage tanks exceeds the maximum water boil off rate of 91 gpm following a complete loss of cooling. The minimum time to boil following a complete loss of cooling at 170 °F exceeds 2 hours. This allows a reasonable time to identify the loss of cooling and initiate makeup water flow to prevent a significant loss of coolant.

Based on the above evaluations, the staff concludes that the proposed revisions to TS 3.9.3 and TS 5.6.3, along with existing operational controls, ensure the available decay heat removal capability will be maintained consistent with its importance to safety, and that the capability to prevent a significant reduction in coolant inventory under accident conditions will be available. Specifically, the decay heat removal capability is acceptable because the SFP cooling system will be capable of maintaining an appropriate pool temperature during planned refueling evolutions; with the failure of a single cooling train, the cooling system will maintain SFP temperature within analyzed limits for SFP structural integrity; and the rack design allows for sufficient natural circulation to maintain the coolant subcooled with the cooling system in operation. Therefore, the proposed revisions to TS 3.9.3 and TS 5.6.3 are acceptable with respect to the resulting increase in decay heat, and the design of the new fuel storage racks provides for acceptable cooling of the stored fuel. The proposed modifications to Surveillance Requirement 4.9.3 and the Bases for TS 3/4.9.3 are consistent with the revised LCO and are, therefore, acceptable.

3.8.2 Conclusion

The existing spent fuel cooling system and associated administrative controls ensure the potential increase in decay heat load resulting from revisions to TS 3/4.9.3 and TS 5.6.3 can be removed with reliability consistent with the importance to safety of decay heat removal. Finally, the existing makeup water supplies provide an adequate capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been published in the *Federal Register* on August 29, 2002 (67 FR 55436). Accordingly, the Commission has determined that the issuance of this amendment will not result in any environmental impacts other than those evaluated in the Final Environmental Impact Statement.

6.0 CONCLUSION

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

REFERENCES

1. Letter from Stephen A. Byrne, SCE&G, to the NRC, “Technical Specification Amendment Request TSP 99-0090 Spent Fuel Pool Storage Expansion,” RC-01-0135, dated July 24, 2001, (Agencywide Documents Access and Management System Accession Number ML012130098).

2. Nuclear Regulatory Commission, Memorandum to Timothy Collins from Laurence Kopp, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel storage at Light Water Reactor Power Plants,” August 19, 1998.

3. American Nuclear Society, “American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants,” ANSI/ANS-57.2-1983, October 7, 1983.

4. NUREG/CR-6683, “A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage,” November 1996 and August 2000.

5. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

6. Virgil C. Summer Nuclear Station, Final Safety Analysis Report.

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TABLE 1 (RADIOLOGICAL CONSEQUENCE ANALYSIS)
ANALYSIS ASSUMPTIONS

Core power, Mwt	2958	
Radial peaking factor	1.7	
Number of damaged fuel assemblies	1.19	
Decay time, hours	72	
Fuel rod gap fractions		
I-131	0.12	
Kr-85	0.30	
All other noble gases, iodines	0.10	
Iodine species fractions		
Elemental	0.9975	
Organic	0.0025	
Pool scrubbing factor		
Elemental	133	
Organic Iodine	1	
Noble Gases	1	
Effective, iodine	100	
Duration of release, hours	2	
Duration of accident, days	30	
Release modeling		
EAB		
FHA in CNMT: 100% release in 2 hours, no filters		
FHA outside CNMT: 100% release in 2 hours, 95% filter		
Control room volume, ft^3	226,040	
CREVS start delay time, minutes		
FHA inside CNMT	10	
FHA outside CNMT	60	
	Before <u>CREVS</u>	After <u>CREVS</u>
CRHE unfiltered makeup flow, cfm	1000	0
CRHE filtered makeup flow, cfm	0	1000
CRHE unfiltered recirculation, cfm	18143	0
CRHE filtered recirculation, cfm	0	18143
CRHE unfiltered in leakage, cfm	10	10
CREVS filter efficiency, %, all species		95
Control room occupancy factors		
0-24 hr		1.0
24-96 hr		0.6
96-720 hr		0.4
Control room breathing rate, m^3/s		3.47E-4
Offsite breathing rate, m^3/s, 0-8 hrs		3.47E-4
Atmospheric dispersion factors, s/m^3		
EAB 0-2 hr		4.08E-4
Control Room 0-8 hr		9.35E-4

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