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SCE&G®
A SCANA COMPANY

October 4, 2006

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

ATTN: Mr. R. E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
REQUEST FOR CORRECTION OF PREVIOUSLY APPROVED OPERATING LICENSE AMENDMENT 160 SAFETY EVALUATION

Reference: Karen R. Cotton, NRC, Letter to Stephen A. Byrne, SCE&G, dated August 30, 2002, "Issuance of Amendment Re: Spent Fuel Pool Expansion (TAC NO. MB2475" - ADAMS Accession No. ML022330203 and ML022470190

South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby resubmits a request for corrections to the Safety Evaluation documenting approval of Amendment 160 to the VCSNS Operating License. The changes are being requested to assure the VCSNS licensing basis remains accurate. The specifics of the request are provided in the attached table.

Should you have questions, please call Mr. Robert G. Sweet at (803) 345-4080.

Very truly yours,

Jeffrey B. Archie

PAR/JBA/dr
Attachments
I. Requested Changes to SE for Amendment 160 to the VCSNS Operating License
II. NRC SER Marked Up Pages
III. VCSNS Documents - Verification Pages

c: K. B. Marsh
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RTS (LAR 02-2891)
File (813.20)
DMS (RC-06-0184)

A001

Requested Changes to SE for Amendment 160 to the VCSNS Operating License

#	NRC SER Page	VCSNS Document Page	➢ Requested Change ➢ Justification
1	Page 2, 1st sentence	N/A	➢ Add the word "storage" between fuel and design. ➢ Amendment 160 permitted the addition of new fuel storage racks into the Spent Fuel Pool. There was no new fuel design analysis performed.
2	Page 2, first paragraph, second sentence	RC-01-0135 Att. VI Page 4-2 Page 4-3 [ML012130115, Pages 2 and 3]	➢ Add parenthetical statement (for accident cases only) when discussing credit for soluble boron. ➢ The LAR submittal stated that the boron concentration will ensure reactivity is maintained less than design limits in the event of a fuel handling accident. The racks are designed to maintain reactivity below design limits during normal storage without the presence of boric acid in the water.
3	Page 2, third paragraph, fourth sentence	N/A	➢ Delete the words "and boration of the water in the spent fuel storage facility". ➢ The water in the Spent Fuel Storage Facility has always been borated. This amendment was not requesting to borate the pool, but to expand the storage capacity of the pool.
4	Page 3, last paragraph, fourth sentence	TS 3.9.1 RC-01-0135 Att. III Page 3 [ML012130098, Page 67]	➢ Revise sentence to " 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} of less than or equal to 0.95 will be maintained in the RCS and refueling cavity in accordance with TS Surveillance Requirement 4.9.1." ➢ TS LCO 3.9.1 allows the determination of either greater than or equal to 2,000 ppm boron concentration or a K_{eff} of less than or equal to 0.95. The additional sentence is directly from the analysis performed for the expansion of storage capability and is the basis for TS LCO 3.7.13.

#	NRC SER Page	VCSNS Document Page	➢ Requested Change ➢ Justification
5	Page 4, first paragraph, second sentence	N/A	➢ Need to add the words "cools the SFP cooling system" and remove the words "removes decay heat from the reactor core when the reactor is in the shutdown condition". ➢ The intent of the LAR was to demonstrate why the expansion of spent fuel storage in the VCSNS Spent Fuel Storage Pool is acceptable. The movement of fuel from the reactor core depends on the temperature of the Component Cooling Water temperature, but being able to ensure all design criteria for the storage of fuel in the pool is maintained also depends on the temperature of the Component Cooling Water.
6	Page 7, Section 3.2.3, first sentence	RC-01-0135 Att. VI Page 4-3 [ML012130115, p3]	➢ Delete the first sentence. ➢ Incorrect statement as VCSNS has accounted for the effect of boric acid in the analysis, but is not taking credit for the presence of boron in the Spent Fuel Pool.
7	Page 12, Section 3.5.1, first paragraph, last full sentence	RC-01-0184 Att. I Page 3 Response to Q-2.a [ML022950156, Page 4]	➢ Delete the words "and wiped down". ➢ VCSNS did not commit to wiping down the old racks being removed from the pool. This is not considered an ALARA practice. Rinsing and surveying the equipment prior to movement away from the pool will ensure that removable contamination is minimized.
8	Page 13, Section 3.5.1, first paragraph, second to the last sentence	RC-01-0135, Att. VI Page 9-6 [ML012130280, Page 111] RC-02-0106, Att. I Page 6 Response to Q-5 [ML021770391, p8]	➢ Delete the words "near the gate slot to the FTC" and replace with "in the first five rows closest to the FTC gate slot." ➢ The analysis performed did not require the entire rack closest to the FTC gate slot to be filled with old fuel. Only the first five rows were required to be filled with old fuel in order to achieve the required dose in the FTC.

#	NRC SER Page	VCSNS Document Page	➢ Requested Change ➢ Justification
9	Page 19, Section 3.7.1, fourth paragraph, second sentence	RC-02-0116, Att. I Page 3 Response to Q-2 [ML021860193, Page 5]	➢ Add the words "at the predetermined lift point" to the end of the sentence, after the words "heavy loads and stored fuel". ➢ There was insufficient storage in the pool to provide 3 feet of empty rack along the entire lift path. There was, however, sufficient storage in the pool to provide 3 feet of empty rack in the surrounding racks at the lift point.
10	Page 25 TABLE 1	N/A	➢ Page 24 was not received by VCSNS. The version on ADAMS has a page -24-, but not a page -25-. This version is the same as page 25.
11	Page 25 TABLE 1	N/A	➢ Release Modeling should state: EAB: FHA in CNMT: 100% release in 2 hours, no filters FHA outside CNMT: 100 % release in 2 hours, 95% filter ➢ To correct data utilized in performing the analysis.
12	Page 25 TABLE 1	RC-02-0089 Page 3 Bottom Row [ML021290133, Page 5]	➢ CRHE recirculation should state: Before CREVS / After CREVS CRHE unfiltered recirculation, cfm: 18143 / 0 CRHE filtered recirculation, cfm: 0 / 18143 ➢ To correct data utilized in performing the analysis.

NRC SER MARKED UP PAGES

1
2
3

 1

2 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.

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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

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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 spent fuel pool 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 will be maintained in the ~~SFP~~ 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. (4)

for a Keff of less than or equal to 0.95

RCS and refueling cavity

Surveillance Requirement

5

5

cools the SF Pcooling system.

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

(6)

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

(6) ~~The licensee also proposes to use a credit for soluble boron to compensate for uncertainties associated with the reactivity equivalencing (i.e., burnup-related) methods.~~ 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.

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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 wiped down~~, and resurveyed by or under the direction of a qualified HPT.

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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

(8)

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 ~~near the gate slot to the FTC~~ in the first five rows closest to the FTC gate slot. Normally, the FTC will be filled with water. (8)

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

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(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.

9 at the predetermined lift point.

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.

(10)

(10) (11) (12)

Note:
Page -24- was not received by VCSNS.

The version on ADAMS has a page -24- but not a page -25-. Their page -24- is the same as our page -26-.
D. Kailey
9-26-02

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

EAB:

FHA in CNMT: 100% release in 2 hours, no filters

FHA outside CNMT: 100 % release in 2 hours, 95% filter

Release modeling

EAB: ~~100% release in 2 hours, via 95% filter~~

(11) ~~Control room for~~ FHA in CNMT: 100% release in 2 hours, no filters

~~Control room for~~ FHA outside CNMT: 100% release in 2 hours, 95% filter

should be ↑

Control room volume, ft^3	226,040
CREVS start delay time, minutes	
FHA inside CNMT	10
FHA outside CNMT	60

	Before CREVS	After CREVS
CRHE unfiltered makeup flow, cfm	1000	0
CRHE filtered makeup flow, cfm	0	1000
(12) CRHE filtered recirculation, cfm	18143	18143
CRHE unfiltered in leakage, cfm	10	10

should be: ↓

	Before CREVS	After CREVS
CRHE unfiltered recirculation, cfm	18143	0
CRHE filtered recirculation, cfm	0	18143

CREVS filter efficiency, %, all species	
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

VCSNS DOCUMENTS

VERIFICATION PAGES

②

- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

USNRC guidelines [4.1.2] and the applicable ANSI standards specify that the maximum effective multiplication factor, k_{eff}, including bias, uncertainties, and calculational statistics, shall be less than or equal to 0.95, with 95% probability at the 95% confidence level.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were employed:

- Moderator is unborated water at a temperature that results in the highest reactivity (4°C, corresponding to the maximum possible moderator density, 1.000 g/cc).
- The racks were assumed to be fully loaded with the most reactive fuel authorized to be stored in the racks.
- No soluble poison (boron) is assumed to be present in the pool water under normal operating conditions, except for the fuel assembly in-transit condition. 2
- Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water.
- The effective multiplication factor of an infinite radial array of fuel assemblies was used in the analyses, except for the assessment of peripheral effects and certain abnormal/accident conditions where neutron leakage is inherent.
- In-core depletion calculations assume conservative operating conditions, highest fuel and moderator temperature, and an allowance for the soluble boron concentrations during in-core operations.
- For assemblies that use WABAs during in-core depletion, it is assumed that the maximum burnup of the assembly when the WABA is removed is 30 GWD/MTU.

The spent fuel storage racks are designed to accommodate the fuel assembly types listed in Table 4.1.1 with a maximum nominal initial enrichment of 4.95 wt% ^{235}U.

Two separate storage regions are provided in the V.C. Summer spent fuel pools. The independent acceptance criteria for storage in each of the regions are as follows:

(2)

⇒ Region 1 is designed to accommodate fresh or burned fuel assemblies with a maximum nominal initial enrichment of 4.95 wt% ^{235}U.

⇒ Region 2 is designed to accommodate fuel assemblies with a maximum nominal initial enrichment of 4.95 wt% ^{235}U which have accumulated a minimum burnup of 41.6 GWD/MTU or fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.1.

The water in the spent fuel storage pool normally contains soluble boron, which would result in a large sub-criticality margin under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 for normal storage be evaluated for the accident condition that assumes the loss of soluble boron. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. Consequences of abnormal and accident conditions have been evaluated, where "abnormal" refers to conditions which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions which are not expected to occur but nevertheless must be protected against. (2)

4.2 Summary of Criticality Analyses

4.2.1 Normal Operating Conditions

The criticality analyses for each of the two separate regions of the spent fuel storage pool are summarized in Tables 4.2.1 and 4.2.2, for the design basis storage conditions. For the fuel acceptance criteria defined in the previous section, the maximum k_{eff} values are shown to be less than 0.95 (95% probability at the 95% confidence level) in each of the regions.

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handling operations. Shipping cask movements will not be performed during the modification period.

Accordingly, the proposed modification does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Spent Fuel Pool have been re-evaluated for the proposed change. The results show that the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, K_{eff} less than or equal to 0.95, will be maintained. The structural damage to the Fuel Handling Building, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed modification, the postulated structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will increase due to the changes in in-core hold time and burnup. The previously calculated doses to thyroid and whole body were 10.6 and 0.52 rem, respectively. The new Exclusion Area Boundary (EAB) thyroid and whole body doses based on the proposed change will be 12.97 and 0.678 rem, respectively. These dose levels will remain "well within" the levels required by 10CFR100, paragraph 11, as defined in Section 15.7.4.II.1 of the Standard Review Plan. Therefore, the increase in dose is not considered a significant increase in consequence.

4

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Reactor Building have also been re-evaluated for the proposed change to assess the affect of higher burnup and shorter cooling time. The proposed re-racking does not affect the fuel assembly mass or drop height parameters. Therefore, the previously determined fuel damage and resulting criticality assessments remain unchanged. However, the radiological dose at the exclusion area boundary will increase due to the changes in in-core hold time and burnup. The previously calculated doses to thyroid were 211 rem. With no action to limit the consequences of the fuel handling accident in the reactor building, the new EAB thyroid dose would be 259 rem. The whole-body would be the same as the doses for the accident in the fuel handling building, since those doses are caused by radionuclides that, in the Fuel-Handling-Building accident, were not affected by the charcoal filters in the building exhaust. This hypothetical thyroid dose would be higher than the criterion of the Standard Review Plan. However, as described in Section 15.4.5.1.4 of the VCSNS FSAR, instrumentation is available to detect the release of radioactivity and to close the Reactor Building Purge System. This action essentially precludes any radioactive release to the environment for this accident. Thus, the results of the postulated fuel drop accidents remain acceptable and do not represent a significant increase in consequences from any of the same previously evaluated accidents that have been reviewed and found acceptable by the NRC.

⇒ Region 1 is designed to accommodate fresh or burned fuel assemblies with a maximum nominal initial enrichment of 4.95 wt% ^{235}U.

⇒ Region 2 is designed to accommodate fuel assemblies with a maximum nominal initial enrichment of 4.95 wt% ^{235}U which have accumulated a minimum burnup of 41.6 GWD/MTU or fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.1.

The water in the spent fuel storage pool normally contains soluble boron, which would result in a large sub-criticality margin under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 for normal storage be evaluated for the accident condition that assumes the loss of soluble boron. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. Consequences of abnormal and accident conditions have been evaluated, where "abnormal" refers to conditions which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions which are not expected to occur but nevertheless must be protected against.

4.2 Summary of Criticality Analyses

4.2.1 Normal Operating Conditions

The criticality analyses for each of the two separate regions of the spent fuel storage pool are summarized in Tables 4.2.1 and 4.2.2, for the design basis storage conditions. For the fuel acceptance criteria defined in the previous section, the maximum k_{eff} values are shown to be less than 0.95 (95% probability at the 95% confidence level) in each of the regions.

Response:

Continuous HP coverage will be provided anytime a diver is in the water, or anything is coming out of the water. Each morning, a pre-job briefing will be held with the HP and the job crew, to ensure that everyone knows what conditions will be present during the day. This briefing will cover Operational Experience pertinent to the day's activities.

2. a. *Describe any radiation surveys that will be performed (from the pool rim or by divers in the pool) to monitor/map dose rates in the SFP, or to check for contamination of material, equipment or divers upon removal from the pool.*

Response:

Radiation surveys shall be completed for all evolutions involved with the reracking operation. All permanently installed items that are to be removed from the pool (such as fuel storage racks, bearing plates, and sparger piping), shall be surveyed prior to removing them from the pool. Additional surveys and either rinsing and/or wiping down will be done as the item breaks the water surface. As a minimum, everything in the pool will be surveyed as it breaks the water surface during removal. 7

Additionally, as applicable, smears shall be taken on items to determine contamination levels and take appropriate measures and controls to minimize the potential for the spread of contamination.

b. *Describe any radiological training (as required by 10 CFR Part 19.12) provided to the divers, specific to the hazards of working around spent fuel, and including the extremity dose hazards of improperly handling (e.g., picking up by hand) potential highly activated debris in the pool.*

Response:

All personnel who obtain unescorted access to the site receive station orientation training (SOT). All individuals involved in the reracking project will receive SOT 2 and 3 which addresses radiation and contamination controls as part of the Plant access training program. Additionally, a project specific training program shall be given to all supporting participants of the project. Specific training regarding the hazards of hot particles, potential for extremity dose hazards due to inappropriate handling of sources, Operating Experience related to rerack projects, etc., will be covered as part of the project specific ALARA briefing that will be conducted prior to project commencement. The ALARA briefing will be incorporated into the pre-job briefing performed daily. The divers will be fully qualified for nuclear diving and will be trained accordingly.

(8)

the rack is completed, the washed section of the rack will be removed from the water, allowing HPs to commence their survey process while the next incremental four feet of the rack are pressure washed. This process continues until the complete rack has been removed from the pool. While the rack hangs above the pool, surveys may show the need to pressure wash the underside of the rack. If necessary, the pressure-washing wand is reconfigured to address the cleaning of the rack underside. Administrative controls will be in place to assure that no water/debris is washed out of the pool

5. *In Section 9.4 of the Holtec Report, page 9-6, shielding calculations in the area near the gate slot to the transfer canal show that to maintain the designated Zone II radiation levels, Rack B-2 (nearest to the gate slot) would need some fuel loading constraints. Describe what constraints will be employed and when they will be instituted to maintain worker doses ALARA. Other than minimizing doses to workers on the crane work platform with the transfer canal empty, when would increased dose rates in this area present a worker dose problem?*

Response:

(8)

The fuel loading constraint required to maintain Zone II radiation levels is for the first 5 rows of Rack B2 to contain no fuel assemblies which were discharged in the last 15 years while the fuel transfer canal (FTC) is empty. Unless required to be drained, the FTC normally remains filled. To administratively control dose rates, the Reactor Engineering procedure which develops spent fuel movement plans will limit the first 5 rows of Rack B2 to fuel with >15 years decay time. The Operations procedure which is used to drain the FTC will notify Health Physics to survey the SFP area and post the radiation zones accordingly. The SFP fuel movement procedure will address the scenario of fuel movement while the FTC is empty. These procedures will ensure that workers in the area are aware of the potential issues, similar to the controls provided in the other radiation zones around the plant. The area most affected by the new spent fuel rack configuration will be within the FTC itself, so any personnel performing fuel transfer system maintenance with the FTC drained will require additional Health Physics controls commensurate with the higher radiation levels. The FTC will remain filled during the rack removal and installation process.

8

The dose rate from stored fuel, at any location above or around the pool, with the pool filled with fuel cooled only 72 hours, would be extremely low – far less than 0.01 mrem/hr. The dose rate to a person on the crane working platform from a 72-hour-cooled fuel assembly in transit at it maximum elevation (as set by the maximum height of the bridge crane hook), will be 2.0 mrem/hr.

The railroad bay is designated as uncontrolled, or Radiation Zone I, dose rate less than 1.0 mrem/hr. An extremely conservative calculation (again, entire SFP filled with fuel assumed to be cooled only 72 hours, a case that is obviously impossible) shows that the combined dose rate from stored fuel plus fuel in transit will be less than the 1.0 mrem/hr limit. For this assessment, fourteen locations were taken as dose rate points in the railroad bay region.

Calculations were performed to determine if old fuel or no fuel would have to be placed in rack locations nearest the gate slot to the transfer canal to limit the dose rate to a person on the crane work platform, with the transfer canal empty. The area is designated Radiation Zone II, dose rate less than 2.5 mrem/hr. The calculations show that old fuel or no fuel placed in the five rows of Rack B2 closest to the gate slot will limit the dose rate from 72-hour-cooled fuel in all other positions in the pool to less than 1.12 mrem/hr. 8

The radionuclide concentrations in the pool water are not expected to increase significantly, for they derive principally from the mixing of primary system water with the pool water and the spalling of crud deposits from the spent fuel assemblies as they are moved in the storage pool during refueling operations. Although the overall capacity of the pool is being increased, the movement of fuel during refueling is independent of storage capacity.

Operating experience has shown that there have been negligible concentrations of airborne radioactivity, and no increases are expected as a result of the expanded storage capacity. Area monitors for airborne activities are available in the immediate vicinity of the spent fuel pool.

No increase in radiation exposure to operating personnel is expected; therefore, neither the current health physics program nor the area monitoring system needs to be modified.

9

(2) Section 3.5 of Attachment V to the amendment request states that racks will not be carried directly over stored fuel. Describe any additional measures that will be used to protect the stored fuel from damage or limit the consequences of such damage. Measures for consideration include establishment of a minimum separation distance between load paths and stored fuel, features that prevent rotation or tipping of the racks should a partial rigging failure occur, limits on load height near stored fuel, and establishment of a minimum decay time for stored fuel prior to rack movement.

Response:

In compliance with NUREG 0612, safe load paths will be included in project specific procedures to ensure that heavy loads shall not be carried over stored fuel in the SFP. Safe load paths will maximize the benefits of strategic fuel shuffles that allow for the greatest distance between a suspended rack and stored fuel while the suspended load is at a height that would allow it to be dropped on stored fuel. A minimum horizontal distance of 3 feet will be maintained between lifted racks and stored fuel. Suspended racks or any other heavy loads that are handled as part of the rerack operation will never be moved over stored fuel assemblies. Additionally, new racks being installed into the SFP will be lowered to a minimal height just above the SFP floor as soon as the rack safely clears the pool perimeter and any pool wall protrusions. As part of the defense-in-depth approach, the action of lowering the rack to a height just above the pool floor prior to commencing any horizontal movement reduces the amount of time that the rack is in a position to do damage to stored spent fuel.

9

All steps involving the handling of heavy loads in and around the spent fuel pool shall be governed and controlled by a project specific procedure. As suggested in NUREG-0612, this procedure will include the safe load paths that will be used for heavy loads traveling over or near the spent fuel pool. Additionally, the procedure will include detailed exhibits showing the rigging configurations for lifting each heavy load. Each rigging exhibit will include the minimum ratings required for each rigging component to comply with NUREG-0612. In general, all steps and quality oversight of the handling of heavy loads will be included as part of the procedure.

Training will be performed with the crew on multiple levels in order to educate them on the many tasks and their associated governing procedures and regulations. Crane operators will get a training session on the functions of the cranes and the new parameters that are introduced by the allowance of travel over the spent fuel pool. In addition to this, and along with the rest of the crew, a training session is given to offer a general overview of the tasks, associated safe load paths, and the applications of NUREG-0612 with respect to the many tasks that will be completed during the project.

(12)

Fuel Handling Accident Input Parameters		
Parameter	FHA Inside Containment	FHA Outside Containment (Inside Fuel Handling Building)
Percent of rod activity in gap, %	I-131 12 KR-85 30 All Others 10	I-131 12 KR-85 30 All Others 10
Iodine composition, % Elemental Organic	 99.75 0.25	 99.75 0.25
Pool decontamination factors Elemental Iodine Organic Iodine Noble gases	 133 1 1	 133 1 1
Purge isolation time, seconds	5.35	NA
Purge flow rate, CFM	20000	NA
Dilution volume above pool, ft^3	33000 (<2% of containment free volume)	NA
Activity released from pool that is released to the environment prior to filtration, %	5.4	100
Charcoal filter exhaust efficiencies, %	0 (No credit taken for reactor building purge exhaust charcoal filters)	95 (Fuel handling building purge exhaust charcoal filters)
Control room habitability envelope (CRHE) volume, ft^3	226040	226040
Control building walls and roof concrete shielding, ft.	2	2
Control room emergency ventilation system (CREVS) actuation.	No credit taken for automatic initiation via RM-A1. Manually start within 10 minutes of the start of the accident.	No credit taken for automatic initiation via RM-A1. Manual start within 60 minutes of the start of the accident
CRHE unfiltered makeup air, CFM	1000 (0 to 10 minutes at which time the CREVS is manually started.) 0 (10 minutes to 30 days)	1000 (0 to 60 minutes at which time the CREVS is manually started.) 0 (60 minutes to 30 days)
CRHE filtered makeup air, CFM	0 (0 to 10 minutes) 1000 (10 minutes to 30 days)	0 (0 to 60 minutes) 1000 (60 minutes to 30 days)
CRHE unfiltered inleakage, CFM	10 (ingress/egress)	10 (ingress/egress)
CRHE filtered recirculation flow, CFM	0 (0 to 10 minutes) 18143 (10 minutes to 30 days)	0 (0 to 60 minutes) 18143 (60 minutes to 30 days)

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