UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

1.0 INTRODUCTION

By letter dated January 23, 1984 South Carolina Electric and Gas Company (the licensee or SCE&G) proposed an amendment to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station.

The amendment would change the spent fuel pool (SFP) storage capacity listed in Technical Specification 5.6 from 682 fuel assemblies to 1276 fuel assemblies in a three region storage design with a maximum initial enrichment for stored fuel of 4.3 weight percent U-235. The nominal center-to-center distance between spent fuel assemblies listed in Technical Specification 5.6 would be changed from 14 inches to slightly greater than 10 inches. A new Technical Specification 3/4.9.12 would be added describing the combination of initial enrichment and cumulative exposure for spent fuel assemblies necessary for storage in Regions 2 and 3. Technical Specifications 5.3.1 and 5.6.1.2 would be changed to reflect storage in the new fuel storage racks of new fuel assemblies for reload enriched up to a maximum of 4.3 weight percent U-235 instead of 3.5 weight percent U-235.

Additional information pertaining to, but not changing, the requested amendment was provided by the licensee (see Section 4.0). As addressed below, the NRC staff has evaluated the safety considerations associated with this amendment. A separate Environmental Assessment addressing this amendment has been prepared.

2.0 EVALUATION

2.1 Criticality Consideration

2.1.1 Region Design

Region 1 design consists of 2 racks, each containing 121 (for a total of 242) stainless steel cells, each of which contains Boraflex absorber on all four sides at an effective boron 10 thickness of 0.022 gm/cm². The cell pitch is 10.40 inches, including a "flux trap" gap of 1.16 inches.

Region 2 has one rack containing 99 stainless steel cells, each containing Boraflex at a thickness of 0.0015 gm/cm² of boron 10. The cell pitch is 10.40 x 10.19 inches with flux trap gaps of 1.26 and 1.05 inches.

Region 3 design consists of 8 racks each containing 121 or 110 stainless steel cells (for a total of 935) which have no fixed boron absorber. The cell pitch is 10.11 inches with a gap of 1.09 inches.

The design is intended to contain standard Westinghouse 17x17 fuel assemblies with an initial enrichment of up to [illegible] percent U-235. Region 1, which can be used, for example, for offloading the core (a full core is 157 assemblies), can accept up to 4.3 percent fuel with no burnup without exceeding criticality limits, and assuming unborated water at a peak (non-accident) reactivity temperature state. Regions 2 and 3 design is intended to contain the same fuel with burnup, with otherwise the same assumptions. To meet the criticality requirements an initial enrichment dependent burnup requirement for the fuel must be met. For Region 2 the burnup must be at least 20,000 MWD/MTU for 4.3 percent enrichment. Lower initial enrichments involve lower burnup, e.g., no burnup for 2.3 percent enrichment. For Region 3, which provides the largest number of storage cells, the burnup required for 4.3 percent (initially) enriched fuel is 42,000 MWD/MTU, dropping to zero burnup for 1.4 percent initial enrichment.

During a core reload in which fuel is to be placed in (burnup) Region 2 or 3, the SCE&G procedures will indicate that the freshly discharged assemblies from the reactor will be initially placed into Region 1. Only after the core has been fully reloaded and the detailed fuel assembly burnup records have been analyzed and verified will fuel be moved into the burnup credit storage locations. This procedure, which has become part of our requirement for multi-region, burnup credit storage pools, is intended to preclude loading errors.

2.1.2 Methods of Calculation

Criticality analyses for these fuel racks were done primarily by Southern Science as a consultant for the rack designer (Joseph Oat Corporation). They have had considerable experience in this area.

The neutron multiplication status of the racks is calculated assuming the fuel cell lattice is infinite in all directions (i.e., no leakage) with pure, unborated water at a temperature of highest reactivity (as derived in a sensitivity study). All of the methods used for calculations are "industry standard" codes which have accumulated considerable experience. The reference criticality analyses of the racks were performed with the AMPX-KENO computer package, using the 123 group GAM-THERMOS cross-section set and NITAWL for U-238 resonance shielding.

For sensitivity calculations (to investigate uncertainties), CASMO, a two-dimensional transport code allowing explicit description of each fuel pin was used. CASMO calculations were compared to AMPX-KENO calculations. Burnup effects were also calculated by CASMO. These burnup results were compared to calculations with EPRI-CELL and NULIF codes. The reactivity effects of axial distribution of burnup were calculated with one-dimensional diffusion theory using CASMO derived diffusion constants.

The base calculation method, AMPX-KENO, has been benchmarked against a number of relevant critical experiments, including those representative of spent fuel racks, by national laboratories, the nuclear industry and by Southern Science. These comparisons have been used to develop biases and variations to use in uncertainty analyses. CASMO has been benchmarked against critical experiments and operating reactors, and in particular comparisons have been made to establish the accuracy of burnup calculations. We conclude that the methods used are state-of-the-art for this problem area and have been suitably verified for these calculations.

Uncertainties

Analyses were done to examine the potential for and magnitude of uncertainties in components of the criticality analyses. The benchmark calculations provided biases and 95/95 (probability/confidence level) uncertainties for the AMPX-KENO base calculations. The tolerance limits for each significant mechanical and material variation of cell and fuel were established and the reactivity effect of these variations was examined. This included variations in B-10 concentration and Boraflex dimensions, box and gap dimensions, fuel enrichment and density and eccentric assembly position in the cell. The investigation provided for the uncertainty of burnup including the effects of non-uniform axial distribution of burnup of the fuel. The axial non-uniformity should in most cases of interest provide a negative contribution compared to a uniform or average burnup (which is the basis for loading a given region), but a positive contribution was included in the uncertainty analysis. We conclude that a suitable range of uncertainty components and reasonable values for these components were used in the analyses.

Results

The results of analyses described above give maximum multiplication (k) for 4.3 percent initial U-235 enrichment for Regions 1, 2 and 3 of 0.941, 0.936 and 0.942, including uncertainty factors (95/95) of 0.0090, 0.0336 and 0.0248 respectively, when using burnups of zero, 20,000, and 42,000 MWD/MTU (the design burnups for these regions). Calculations were then done (iteratively) to obtain the same k values for other initial enrichments thus providing a burnup versus enrichment function for Regions 2 and 3. The results meet the NRC criterion for spent fuel peak for k less than 0.95 (for unborated water) including uncertainties.

The k results were determined with water temperatures within the non-accident range giving maximum reactivities (i.e., 40°, 68° and 150°F for Regions 1, 2 and 3, respectively). For Region 3 the peak k occurs at 248°F (boiling state for the pool at fuel level) which is considered an accident condition. However, at this temperature k, with uncertainty included, still does not exceed 0.95, and additional voiding in the cell introduces negative reactivity. Thus the results are satisfactory for peak pool temperatures.

The results do not take credit for the long term changes in reactivity (primarily from Pu-241 decay) which can introduce significant negative reactivity. For example, the change for 42,000 MWD/MTU fuel is calculated to be over - 1% Δk for 1 year and - 6% Δk for 10 years storage time.

The strategy of spent fuel storage used in this design requires the fuel to accumulate considerable burnup in order to fully utilize the storage capacity, particularly for Region 3 which constitutes the bulk of the cells. An examination of the history (and expectations) of discharge burnup versus enrichment for many reactors, including more recent higher burnup fuel examples, indicates that generally all of the examined discharges fall well above the Region 3 "acceptable" curve of Figure 2. Thus the expectation is that there will be no problem meeting the burnup requirements for storage of a given enrichment batch. Should unusual circumstances, however, cause some discharged fuel to fail to meet the required burnup, long term storage (in Region 3) could be accommodated via a system of "checkerboarding" fuel or possibly by taking advantage of arrangements taking credit for long term decay effects. However, these systems are not part of this submittal or review and would require separate design, calculation and review considerations as well as further licensing authorization should the need arise.

2.1.3 Accident Analyses

The reactivity effects of postulated abnormal or accident conditions has been considered. These include misloading an assembly into an incorrect region, pool temperature variations, dropped fuel assembly during pool loading, fuel assembly outside of the racks and rack movement from seismic conditions. None of these conditions result in exceeding the limiting k criteria of 0.95. Procedures exist to assure that assemblies discharged from the core are moved only into Region 1 which can safely accommodate even fresh fuel of 4.3 percent enrichment. Movement to other regions will occur only after burnup records are analyzed and verified and suitable region assignment determined. Thus administrative procedures are established to preclude the misloading event, but if it occurs, the pool borated water (2000 ppm or greater) will maintain k below 0.95. If the licensee's analysis did not take credit for this boron, NRC review policy permits credit for this boron for the event condition which would involve two unlikely unrelated concurrent events (e.g., misloading and loss of pool water boron).

The region reactivity calculations were done for maximum, non-abnormal, pool temperatures, but even when normal temperature ranges are exceeded k does not exceed 0.95. The dropped assembly does not approach the racks closer than 12 inches and thus remains effectively isolated from them and does not increase k. The region outside the racks is limited and prevents assemblies adjacent to the racks except in a "reserved area" for which administrative control should prevent assembly placement. As with mis-loading, the pool water boron would maintain k under 0.95 should an error result in an assembly in this area adjacent to the racks. Lateral movement of the racks under seismic conditions has been examined and motion is not sufficient to decrease spacing significantly.

We conclude that a suitable range of events has been examined and the results are within our criteria and are satisfactory.

2.1.4 New Fuel Racks

SCE&G has also analyzed the criticality aspects of the storage of 4.3 percent enriched fresh fuel in the (dry) new fuel storage racks. These are two arrays of 30 0.075 inch stainless steel box cells, with a mininum of 21 inches between cell centers. These have been analyzed over a full range of assumed moderator (unborated) density, using AMPX-KENO for low density and diffusion theory for high density. These calculations indicate that k remains (well) below 0.95 for the entire density range (maximum is 0.915 at high density, 0.82 at low density), and would do so including uncertainties.

2.1.5 Summary and Conclusions

SCE&G has proposed for Summer a spent fuel rack design for storage of Westinghouse standard 17x17 fuel with enrichments up to 4.3 percent U-235. The design incorporates three regions, each with sufficient fixed boron to meet criticality criteria for a specific range of enrichments and burnup combinations. We have reviewed and accepted several spent fuel pool designs involving multiple storage regions and taking credit for burnup in a similar manner (e.g., SNUPPS reactors, Arkansas Nuclear One Unit 1, Fort Calhoun). The Summer criticality analysis has been done with state-of-the-art methods which have been verified by comparison with experiment and are acceptable. Conservative assumptions have been made about the fuel and rack material and dimensional conditions and the pool conditions. Suitable uncertainties have been considered on determining the multiplication status. Suitable procedures have been developed to minimize misplacement events. Credible accidents and conditions have been considered. The various effective multiplication factors meet our acceptance criteria.

We thus conclude that the proposed design of the spent fuel pool is suitable as regards criticality, and General Design Criterion 62 is satisfied. The design of the new fuel racks for 4.3 percent enriched fuel of the same design is also suitable.

2.2 Spent Fuel Pool Cooling and Makeup

2.2.1 Evaluation

The existing spent fuel pool cooling system consists of two independent spent fuel pool cooling loops each with a 1,800 gpm pump, heat exchanger, valves and instrumentation. The two loops are interconnected such that it is possible to bypass either pump or heat exchanger should it be powered from separate emergency (diesel) power sources. Each loop is rated for the removal of 14.02×10^6 BTU/hr when the pool water temperature is 135°F and the component cooling water inlet temperature to the heat exchanger is at 105°F.

The licensee indicates the maximum pool water temperature will be 140°F for the maximum normal heat load when only one cooling train is in operation. Similarly the maximum pool water temperature will be 139°F for the maximum abnormal heat load when both cooling trains are in operation. The assumptions made in establishing the pool water temperature for the above two heat loads are in accordance with NUREG-0800, "Standard Review Plan" (SRP) Section 9.1.3. We have independently calculated the bulk pool water temperature based on our calculated decay heat loads. The results are in close agreement with that presented in the licensee's submittal. Since the maximum water temperature for the maximum normal heat load does not exceed 140°F and the temperature of the maximum abnormal heat load is less than the boiling temperature, we conclude the existing spent fuel pool cooling system is acceptable for the new calculated heat loads.

SCE&G has also calculated the time to boil for the maximum normal and abnormal heat loads. The calculations assume that all pool cooling is lost and that no heat is transferred to the pool walls and abnormal heat loads. The boiloff rates with these assumptions were calculated by the licensee to be 33.8 gpm and 64.6 gpm, respectively. For the maximum normal and abnormal heat loads, the pool boiling temperature would not be reached for approximately ten and five hours, respectively. Our independent calculations are in close agreement with the above values. We conclude that these time intervals provide reasonable assurance that corrective actions can be completed before boiling would occur.

The licensee indicates there are three discrete sources of makeup water to provide assurance that the stored spent fuel will not be uncovered in the event all cooling is lost. They are the demineralized water storage tank, the refueling water storage tank, and the reactor makeup water storage tank. Normally, pool makeup is capable of being provided by the demineralized water storage tank at a rate of 65 gpm. While this tank is not capable of withstanding the SSE, two seismic Category 1 backup makeup water sources are available (the refueling water storage tank and the reactor makeup water storage tank) and each is capable of supplying makeup water at a rate of 65 gpm. The above makeup rates are in excess of the calculated boiloff rates. We, therefore, conclude that reasonable assurance has been provided that makeup water in sufficient quantity has been provided and the makeup capability is therefore acceptable.

The margin between the temperature of water exiting from the rack storage cell and the pool bulk water temperature was established by the licensee assuming the most choked flow location. It was found that the maximum temperature exiting from a storage cell would be 170°F while the corresponding saturation temperature would be 240°F. We conclude that this difference in these two temperatures provides sufficient assurance that local boiling will not occur and the rack designs are therefore acceptable in this regard.

2.2.2 Conclusion

Based on our review of the licensee's submittals, we conclude the following:

1. The calculated maximum normal and abnormal heat load values are consistent with those we calculated using the guidance of SRP Section 9.1.3 and are, therefore, acceptable.

2. Using our calculated maximum normal heat load, we have confirmed that the pool water temperature will not exceed 140°F when only one spent fuel pool cooling loop is in operation, and the spent fuel pool cooling system is, therefore, acceptable.

3. Using our calculated maximum abnormal heat load, we have confirmed that the pool water temperature will be below boiling when both spent fuel pool cooling loops are in operation and the spent fuel pool cooling system is, therefore, acceptable.

4. The time interval before boiling assuming the loss of all cooling for the maximum normal and abnormal heat loads will be 10 hours and 5 hours, respectively. From this we conclude that sufficient time is available to restore pool cooling before boiling occurs.

5. The boil-off rate for the maximum normal and abnormal heat load conditions would be 34 gpm and 65 gpm. Considering that there are three sources of makeup water, two of which are seismic Category I, we conclude that reasonable assurance has been provided that the fuel will not be uncovered in the event boiling were to occur.

6. Sufficient margin exists between the storage cell water exit temperature and the corresponding saturation temperature to assure that local boiling will not occur.

Therefore, we conclude that the spent fuel pool cooling and makeup system is acceptable.

2.3 Installation of Racks and Load Handling

2.3.1 Evaluation

SCE&G proposes to replace the existing empty racks before the first refueling with eleven free standing high density storage racks that have been fabricated by Joseph Oat Corporation from ASTM 240-304 stainless steel sheet and plate. The nominal interior dimensions of all storage cells will be 8.85 x 8.85 inches. A minimum distance of 1 7/8 inches will be maintained between storage racks. This modification will increase the pool storage capacity from 682 to 1276 fuel assemblies.

Of the 1276 storage cells, 242 will be provided in two Region I type storage racks, each having a 11 x 11 array of storage cells. There will be 99 storage cells provided in one Region II type storage rack having a 9 x 11 array of storage cells. Further, there will be a total of 935 Region III storage cells provided in eight Region III type storage racks. Five of these storage racks will have a 11 x 11 array of storage cells and the remaining three storage racks will have a 11 x 10 array of storage cells.

The storage racks will be designed, constructed and assembled in accordance with AISC Manual of Steel Construction, ANSI N210-1976 Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power, ASME Section III, ASNT-TC lA, ASTM-A240, ASME Section II and AMSE Section IX.

The V. C. Summer spent fuel pool currently does not contain stored spent fuel assemblies and SCE&G states the reracking will be completed before the first refueling occurs. Therefore, the potential for a load drop on spent fuel assemblies does not exist and a potential radiological release from load drops during reracking is not a concern.

Since the range of travel of the fuel handling building crane will not permit the crane hook to be centered over the storage racks, a temporary gantry crane will be installed for the removal of the existing storage racks and the installation of the new storage racks. This crane will have a rated capacity of twenty tons compared to the weight of the heaviest storage rack of 18.15 tons. It has been designed in accordance with CMAA-7D and load tested in accordance with ANSI B30.2.

2.3.2 Conclusion

We have reviewed the licensee's proposed amendment with regard to rack installation and load handling and conclude that it is acceptable.

2.4 Structural Design

2.4.1 Introduction

The high density rack modules for long term fuel storage are located in the spent fuel pool of the fuel handling building. The spent fuel pool structure is a reinforced concrete structure supported on caissons down to competent rock and is integrated with the remainder of the building. The pool walls and slab are 6'-0" thick and the caissons are 3'-0" and 4'-0" in diameter.

The new racks are stainless steel "egg-crate" structures. These cells are supported on a heavily welded base. The racks are each free-standing on the pool floor.

2.4.2 Applicable Codes, Standards and Specifications

Load combinations and acceptance criteria were compared with those found in the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, and amended January 18, 1979. The existing concrete pool structure was evaluated for the new loads in accordance with the requirements of NUREG-0800, Standard Review Plan Section 3.8.4 and the Summer FSAR.

2.4.3 Loads and Load Combinations

Loads and load combinations for the racks and the pool structure were reviewed and found to be in agreement with the applicable portions of the NRC Position.

2.4.4 Seismic and Impact Loads

Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. The seismic loads were applied to the model in three orthogonal directions simultaneously. Damping values for the seismic analysis of the racks and the pool structure were taken as 2 percent for the Operating Basis Earthquake (OBE) and 4 percent for the safe shutdown earthquake (SSE). Rack/fuel bundle interactions were considered in the structural analysis.

2.4.5 Design and Analysis Procedures

a. Design and Analysis of the Racks

A non-linear 3-dimensional time-history analysis of the rack module model was performed. The model included mass, spring, damping, and gap elements and accounts for sliding, tipping and potential rack-to-rack interaction in order to determine stresses and strains within the racks. Partial as well as fully loaded racks were analyzed with range of sliding friction coefficients, between 0.8 and 0.2.

Calculated stresses for the racks components were found to be well within allowable limit. The racks were found to have adequate margins against sliding and tipping.

An analysis was conducted to assess the potential effects of a dropped fuel bundle on the racks and results were considered satisfactory.

An analysis was conducted to assess the potential effects of a stuck fuel assembly causing an uplift load on the racks and a corresponding downward load on the lifting device as well as a tension in the fuel assembly. Resulting stresses were found to be within acceptance limits.

b. Analysis of the Pool Structure

The Summer fuel pool is a reinforced concrete structure. The walls and slab were analyzed as a continuous two-dimensional frame supported by the caissons and surrounding intergral concrete floors. The design of the caissons was reevaluated to include the effects of the additional loads from the new racks and spent fuel.

The capacity of the walls and the slab is dependent upon the interaction curve of bending moments and membrane forces. The results show that the walls and slab have sufficient capacity to sustain the loading from the new rack conditions. The capacity of the caisson is dependent upon the interaction curve of bending moments and axial forces. The results show that all the affected caissons have sufficient capacity to sustain the loading for the new rack conditions with further margin available.

2.4.6 Conclusion

It is concluded that the proposed rack installation will satisfy the requirements of 10 CFR 50, Appendix A, GDC 2, 4, 61 and 62 as applicable to structures and therefore, is acceptable.

2.5 Materials

2.5.1 Description

The safety function of the spent fuel pool and storage rack system is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions.

The spent fuel racks in the proposed expansion would be constructed of Type 304 stainless steel, except for the nuclear poison material and the material of the adjustable supports. The adjustable supports are fabricated with Alloy-Nitronic-60 to reduce galling. The existing spent fuel pool liner is constructed of stainless steel. The high density spent fuel storage racks will utilize Boraflex[1] sheets as a neutron absorber. Boraflex consists of boron carbide powder in a rubber like silicone polymeric matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure. The major components of the assembly are the fuel assembly cells, the Boraflex material, and lower spacer plates with the adjustable supports.

The upper end of the cell has a funnel shape flare for easy insertion of the fuel assembly. An angular structural element surrounds the Boraflex material, but is open at the top and bottom to provide for venting of any gases that area generated. The Boraflex sheet sits in a square annular cavity formed by the square inner stainless steel tube and the outer angular element.

The pool contains oxygen-saturated demineralized water containing boric acid. The water chemistry control of the spent fuel pool has been reviewed elsewhere and found to meet NRC recommendations.

2.5.2 Evaluation

We have reviewed the compatibility and chemical stability of the materials, except the fuel assemblies, wetted by the pool water. The pool liner, rack lattice structure and fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the Type 304 stainless steel should exceed a depth of 6.00×10^{-5} inches in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. The Boraflex is composed of non-metallic materials and therefore will not develop a galvanic potential in contact with metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material. The evaluation tests have shown that the Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan,[2] exposing Boraflex to 1.103×10^{11}

rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boraflex. Long term borated water soak tests at high temperatures were also conducted.[3] The tests show that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The annulus space which contains the Boraflex is vented to the pool at each storage tube assembly. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the inner stainless steel tube.

The tests[1] have shown that neither irradiation, environment nor Boraflex composition has a discernible effect on the neutron transmission of the Boraflex material. The tests also show that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally in the Boraflex will typically contain 0.1 wt percent of soluble boron. the test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble specie from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically.

2.5.3 Conclusion

From our evaluation as discussed above we conclude that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life.

We further conclude that the environmental compatibility and stability of the materials used in the expanded spent fuel storage pool is adequate based on the test data cited above and actual service experience in operating reactors.

We have reviewed the surveillance program and we conclude that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. The materials surveillance program spelled out by the licensee will reveal any instances of deterioration of the Boraflex that might lead to the loss of neutron absorbing power during the life of the new spent fuel racks. We do not anticipate that such deterioration will occur. This monitoring program will ensure that, in the unlikely situaation that the Boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

We, therefore, find that the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison and is, therefore, acceptable.

References

1. J. S. Anderson, "Boraflex Neutron Shielding Material. Product Performance Date," Branch Industries, Inc., Report 748-30-1, (August 1979).

2. J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1, (August 1981).

3. J. S. Anderson, "A Final Report on the Effects of High Temperature Borated Water Exposure on BISCO Boraflex Neutron Absorbing Materials," Brand Industries, Inc., Report 748-21-1, (August 1978).

2.6 Spent Fuel Pool Cleanup System

2.6.1 Evaluation

The SFP cleanup system is part of the pool cooling system. It consists of a bypass flow (180 gpm) that passes through an ion exchange demineralizer followed by a cartridge type filter. There is also a separate skimmer system to remove surface dust and debris from the SFP. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at acceptably low levels.

The proposed modification will result in only a small increase in radioactivity released to the spent fuel pool. The existing SFP cleanup system is capable of handling this small increase and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

2.7 Occupational Radiation Exposure

2.7.1 Evaluation

The staff has evaluated the radiation protection aspects of the licensee's plans to modify the spent fuel pool as described in Chapter 8, "Environmental Evaluation" of the report entitled, "V. C. Summer High Density Spent Fuel Storage Racks." This report was submitted by SCE&G in support of the amendment for the installation of the high density spent fuel storage racks.

The basis of our acceptance of Summer's occupational dose control programs is that doses to personnel will be maintained within the limits of 10 CFR 20 "Standard for Protection Against Radiation," and as low as is reasonably achievable (ALARA).

The spent fuel pool at Summer has never been used to store irradiated fuel assemblies and contains only a minimal amount of contamination. Radiation levels have been measured at three (3) depths within the pool and a maximum exposure rate of 0.5 mR/hr has been detected at the bottom of the pool. Because of the low exposure rates, personnel exposure is expected to be minimal. However, the licensee has taken measures to ensure that personnel exposures to divers working in the spent fuel pool are ALARA. These measures include:

(1) Reviewing all procedures for removing and installing the racks with the diving contractor,

(2) All work will be done under the radiation work permit (RWP) program to ensure that doses are ALARA,

(3) All divers will be issued personnel dosimetry and any doses received will be carefully monitored,

(4) Vacuums will be used to clear the floors of the spent fuel pool after the removal of the old racks.

The licensee does not expect any significant increase in radiation levels due to the buildup of radioactive crud along the side of the pool. If crud buildup eventually becomes a major contributor to pool radiation levels, measures will be taken to reduce such exposure rates. The purification system for the pool includes filters and demineralizers to remove crud and will be operating during the modification of the pool.

The licensee performed a three-dimensional shielding analysis on the spent fuel pool assuming the pool is filled to capacity with the proposed storage densification arrangement. This analysis shows that radiation exposure rates will be less than 1 mR/hr on the outside of the pool walls and at the pool surface from the stored spent fuel. This radiation level meets the V. C. Summer design radiation zoning for the fuel handling building. The shielding analysis was performed using the shielding codes recommended by the staff in NUREG-0800 and, therefore, is acceptable.

SCE&G has presented the following plans for the removal and disposal of the existing racks. The present racks will be unbolted and removed from the pool by divers using a temporarily installed crane. The old racks will receive an initial high pressure water spray in the decontamination pit to remove the majority of the surface contamination. The exposure rate from this surface contamination is estimated to be 0.5 mR/hr. The racks will be temporarily stored in the fuel handling building. SCE&G is considering several options for removing the racks which include: contractor removal, in-house decontamination and disposal, and in-house decontamination and storage on site for possible future use. The staff will monitor the final disposals of these racks.

We have estimated the increment in occupational dose during normal operations, after the pool modification, resulting from the proposed increase in stored fuel assemblies. The spent fuel assemblies contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel; the major source of exposure is the radionuclide concentrations in the pool water. The most significant contributor to the radionuclides is the movement of fuel rather than the number of fuel assemblies in the pool. Thus the additional assemblies will add a negligible amount to area dose rates. Based on present and projected operations in the spent fuel pool area and experience from similar modifications, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation dose to plant personnel. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational doses within the limits of 10 CFR Part 20, and ALARA.

2.7.2 Conclusion

Based on our review of the Summer SFP modification description and relevant experience from other operating reactors that have performed similar modifications, the staff concludes that the licensee's modification can be performed within the limits of 10 CFR Part 20 and in a manner that will maintain doses to workers ALARA, and therefore, is acceptable.

2.8 Radioactive Waste Treatment

2.8.1 Evaluation

The plant contains waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation, dated February 1981. There will be no change in the waste treatment system or in the conclusions given in Sections 9.0 and 11.0 of the evaluation of these systems because of the proposed modification. Our evaluation of the SFP cleanup system, in light of the proposed modification, has concluded that any resultant additional burden on the system is minimal because the added fuel would contribute little or no additional radioactivity. Therefore, the existing SFP cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water within acceptably low levels.

2.8.2 Conclusion

Our evaluation of the radiological considerations supports the conclusion that the proposed modification to the spent fuel pool at Virgil C. Summer is acceptable because:

(1) The conclusions of the evaluation of the waste treatment systems, as found in the Virgil C. Summer Safety Evaluation Report (February 1981), are unchanged by the modification.

(2) The existing spent fuel pool cleanup system is adequate for the proposed modification.

2.9 Radiological Consequences of Cask Drop and Fuel Handling Accidents

2.9.1 Evaluation

Two accident types were considered (a cask drop and fuel assembly drop) to characterize the radiological consequences of incidents involving the spent fuel pool as discussed below.

Cask Drop Accident

The licensee has stated that the spent fuel cask will not be lifted more than 30 ft. above an unyielding surface (except over the flooded cask loading pit which is effectively equivalent to a 30 ft. drop in air) during the entire transfer operation under normal operating conditions. On this basis, no radiological release is anticipated from such a drop, and, therefore, no doses need be evaluated in accordance with Standard Review Plan 15.7.5.

Fuel Handling Accident

For a fuel handling accident, it is assumed that a fuel assembly is dropped by the refueling crane into the reactor core or spent fuel pool. The licensee has proposed to expand the storage capacity of the SFP from 682 spent fuel assemblies to 1276 assemblies which require a re-evaluation of the fuel handling accident presented in the SER issued February 1981.

The new high density racks will be installed prior to the first refueling outage; the spent fuel pool contains no spent fuel at this time. Although the high density racks are designed for high burnup fuel for possible future use, the staff notes that a burnup of 38,000 MWD/MTU is the limit at this time. Therefore, the staff's review indicates that the proposed spent fuel pool modification does not increase radiological consequences of fuel handling accidents considered in the staff Safety Evaluation of February 1981, since this accident would still result in, at most, release of the gap activity of one fuel assembly due to the limitation on available impact kinetic energy.

2.9.2 Conclusion

The above accident evaluations are based on the criteria contained in Standard Review Plans 15.7.4 and 15.7.5 and Regulatory Guide 1.25. Based on this review, the staff concludes that the radiological consequences of this proposal are acceptable.

CONCLUSION

3.0 The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (49 FR 26846) on June 29, 1984, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 References

All letters from O. W. Dixon, Jr. (SCE&G) to H. R. Denton (NRC):
January 23, 1984
March 6, 1984
April 4, 1984
April 17, 1984
May 11, 1984
May 18, 1984
May 30, 1984
July 10, 1984
July 31, 1984
July 31, 1984
August 8, 1984
August 17, 1984

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Dated: September 27, 1984

September 27, 1984

AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF-12 - Virgil C. Summer Unit 1

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