December 15, 1983

Docket No. 50-325
and 50-324

Mr. E. E. Utley
Executive Vice President
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment Nos. 61 and 87 to Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications and are in response to your letter dated April 16, 1981 supplemented by letters dated June 22 and November 23, 1981, March 16, April 5, May 20, and September 16, 1982, February 23, March 31, May 5 and September 20, 1983.

These amendments allow an increase in the Brunswick 1 and 2 spent fuel storage capacity to a maximum of 3946 assemblies.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and Notice of Issuance are also enclosed.

Sincerely,

Original signed by/

Marshall Grotenhuis, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-71
2. Amendment No. 87 to DPR-62
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice

cc w/enclosures:
See next page

DISTRIBUTION:	Docket File	NRC PDR	LPDR	ORB#2 Reading	DEisenhut
SNorris	MGrotenhuis	OELD	SECY	LJHarmon	EJordan
JTaylor	TBarnhart-4	WJones	DBrinkman	ACRS-10	OPA-CMiles
RDiggs	NSIC	Gray	Extra - 5		

DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DL:AD:OR
SNorris	MGrotenhuis	DVassallo	CSard	GLainas
12/6 /83	12/6 /83	12/ /83	12/14/83	12/14/83

Mr. E. E. Utley
Carolina Power & Light Company
Brunswick Steam Electric Plant, Units 1 and 2

cc:

Richard E. Jones, Esquire
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

George F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. Charles R. Dietz
Plant Manager
Post Office Box 458
Southport, North Carolina 28461

Mr. Franky Thomas, Chairman
Board of Commissioners
Post Office Box 249
Bolivia, North Carolina 28422

Mrs. Chrys Baggett
State Clearinghouse
Budget and Management
116 West Jones Street
Raleigh, North Carolina 27603

U. S. Environmental Protection
Agency
Region IV Office
Regional Radiation Representative
345 Courtland Street, N. W.
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
Star Route 1
Post Office Box 208
Southport, North Carolina 28461

James P. O'Reilly
Regional Administrator
Region II Office
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Dayne H. Browns, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
Post Office Box 12200
Raleigh, North Carolina 27605

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. DPR-71

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated April 16, 1981 as supplemented by letters dated June 22 and November 23, 1981, March 16, April 5, May 20, and September 16, 1982, February 23, March 31, May 5 and September 20, 1983 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: December 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. [illegible]

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following page of the Appendix A Technical Specifications with the enclosed page as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
5-5	5-5

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The new fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to less than 0.90 when dry and less than 0.95 when flooded with unborated water.

5.6.1.2 The spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to less than 0.95 with the storage pool filled with unborated water with:

a. New PWR fuel containing not more than 41 grams of U-235 per axial centimeter of active fuel assembly, and a maximum assembly average loading of 3.2 w/o U-235.

b. New BWR fuel containing not more than 15.6 grams of U-235 per axial centimeter of active fuel assembly, and a maximum assembly average loading of 3.0 w/o U-235.

5.1.6.3 The k_{eff} for the unpoisoned racks includes a conservative allowance of 0.5% $\Delta k/k$ for uncertainties. The k_{eff} calculated for the poisoned racks includes the sum of all appropriate biases and the root-mean-square (RMS) of the uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 116'4".

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 160 PWR fuel assemblies and 1803 BWR fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cycle or transient limits of Table 5.7.1-1.

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated April 16, 1981 as supplemented by letters dated June 22 and November 23, 1981, March 16, April 5, May 20, and September 16, 1982, February 23, March 31, May 5 and September 20, 1983 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 87 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: December 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following page of the Appendix A Technical Specifications with the enclosed page as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
5-5	5-5

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The new fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to less than 0.90 when dry and less than 0.95 when flooded with unborated water.

5.6.1.2 The spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to less than 0.95 with the storage pool filled with unborated water with:

a. New PWR fuel containing not more than 41 grams of U-235 per axial centimeter of active fuel assembly, and a maximum assembly average loading of 3.2 w/o U-235.

b. New BWR fuel containing not more than 15.6 grams of U-235 per axial centimeter of active fuel assembly, and a maximum assembly average loading of 3.0 w/o U-235.

5.1.6.3 The k_{eff} for the unpoisoned racks includes a conservative allowance of 0.5% $\Delta k/k$ for uncertainties. The k_{eff} calculated for the poisoned racks includes the sum of all appropriate biases and the root-mean-square (RMS) of the uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 116'4".

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 144 PWR fuel assemblies and 1839 BWR fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cycle or transient limits of Table 5.7.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 61 TO FACILITY LICENSE NO. DPR-71 AND
AMENDMENT NO. 87 TO FACILITY LICENSE NO. DPR-62
CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated April 16, 1981 and supplemented by letters dated June 22, and November 23, 1981, March 16, April 5, May 20, and September 16, 1982, February 23, March 31, May 5, and September 20, 1983, Carolina Power & Light Company (CP&L, the licensee) requested amendments to Facility Operating Licenses DPR-71 and DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The request is to authorize increased storage capability for boiling water reactor (BWR) fuel in the spent fuel pools (SFP) for the two nuclear units. The authorized storage capability for pressurized water reactor (PWR) fuel would be decreased. The proposed modifications would change the SFP storage spaces from 616 PWR or 1386 BWR spaces per unit to 160 PWR and 1803 BWR licensed spaces for Unit 1 and 144 PWR and 1839 BWR licensed spaces for Unit 2. This expanded storage capacity will allow the continued operation of the two nuclear units with onsite storage of spent fuel to 1988 for Unit 1 and 1987 for Unit 2. This expanded capacity would provide full core discharge capability until the stated times.

The licensee's proposal would increase the SFP storage capacity by replacing some of the existing spent fuel storage racks with new high density storage racks. The new racks will contain neutron absorber material in the rack walls so that spacing between stored assemblies can be reduced while maintaining adequate criticality margin. 1173 new spaces in each fuel pool are contained in high density racks made up of modules, each module being composed of six-inch square cells, each cell accommodating a single BWR fuel assembly. The cell walls contain a neutron absorber material sandwiched between sheets of stainless steel. The cells making up the module have 6.56-inch center-to-center spacing. The spacing is sufficient to maintain K_{eff} below 0.95. The racks are also designed in such a manner that accidental dropping of a fuel assembly will not cause a geometry that could result in criticality.

The licensee's basic supporting document for this action is a report, Spent Fuel Storage Expansion Report, that was attached to their submittal dated April 16, 1981. The report contains an overall description of the racks, their design bases, a description of the proposed installation as well as the licensee's analyses and evaluations supporting the proposed spent fuel pool expansion.

The staff evaluation of the safety considerations associated with this proposed action are addressed below. A separate Environmental Impact Appraisal has been prepared for this action.

2.0 DISCUSSION AND EVALUATION

2.1 Structural and Mechanical Design Considerations

Description of the Spent Fuel Pool and Racks

Both units at Brunswick are Mark I BWRs. The spent fuel pools are right and left hand and symmetrical with respect to the transverse centerline of the plant. The pools are elevated with the top of the pools at the fueling floor level, elevation 117'4". Grade is at elevation 19'6". The inside dimensions of the pools are:

height = 38.75'
length = 46'
width = approximately 28'

The pool structures are reinforced concrete with floor thickness of about 5.5' and walls of various thickness from 4 to 5'. The ends of each pool (column lines N and P) are a portion of two prestressed concrete girders which run the length of each reactor building. Each of these girders is 140' long by 5' wide by 42.33' deep. The girders support the spent fuel pool, reactor well, steam separator well, and portions of the floor slabs at elevations 80'0", 98'8", and 117'4". The ends of the girders are supported by the exterior reactor building walls and are independent of the reactor containments.

Each pool is lined with a continuous, welded, watertight, 1/4" thick stainless steel plate which is backed up by leak-chase channels at all seams.

It is proposed to remove existing racks at the end of each pool and replace them with high density racks. Existing racks in the center portions of each pool will remain in place.

The existing racks are fabricated of stainless steel and are supported within an "egg-crate" grid of trusses which rests on the pool floors. Individual existing racks will be removed from the grid of trusses in order to make room for the new racks and the grid is to remain in place. The new racks (also stainless steel) are to be installed over the existing grid on a system of free-standing pedestals which will allow the free-standing racks to bridge the grid. Leveling of the racks will be accomplished with shims, if necessary.

The new racks are stainless steel "egg-crate" structures. The 15 by 17 cell rack is about 8.33' wide by about 9.33' long by about 14.5' high. The pedestals mentioned above are heavy stainless steel plate, the largest of which is about 3.33' long by about 2' wide by about 1.8' high. The pedestals are placed at the corners of the racks and several of them support adjacent corners of two racks. The pedestals are constrained by friction.

Evaluation

Applicable Codes, Standards and Specifications

Structural material of the racks conforms to the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NF. Computed stresses were compared with the ASME B&PV Code, Section III, Subsection NF. Load combinations and acceptance criteria for the racks are in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979 (hereafter referred to as the "NRC Position"). Buckling criteria for the cold formed stainless steel portions of the racks were in accordance with the AISI "Cold Formed Stainless Steel Structural Design Manual", 1979 edition.

The pool structure was evaluated in accordance with the requirements of ACI-318-71 which is found acceptable.

Load and Load Combinations

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

Seismic and Impact Loads

Seismic loads are based on the original design floor acceleration response spectra calculated at elevation 80'0" of the plant. This was based on a 0.16 g. DBE with 4 percent structural damping and 1 percent equipment damping. Acceleration in the vertical direction was computed as being 2/3 of horizontal acceleration. Impact loads due to rack/fuel bundle interaction were considered.

Loads due to a fuel bundle drop accident were considered in a separate analysis for such an occurrence.

The postulated loads from such events were found to be acceptable.

Design and Analysis of the Racks

Artificial earthquake time histories were generated from the floor response spectra and used as input to a 2 dimensional stick model of the racks. The racks were found to be essentially rigid in the vertical direction for purposes of structural design. Effects of added hydrodynamic effects due to motion of the racks in the water were accounted for. A separate analysis using a simplified model and a constant 1 g acceleration was used to assess the effects of rack/fuel bundle impact on the rack structural design.

An analysis for sliding and tipping of the racks was accomplished. It was found that the racks are placed sufficiently far apart to preclude the possibility of rack-to-rack or rack-to-pool structural interaction. The racks and pedestals were found to be stable for all postulated events.

The rack structural design produced calculated stresses for the rack components which were well within allowable limits. The pedestals were designed

for worst-case loading. It was found that an analysis was conducted to assess the potential effects of a dropped fuel bundle on the racks and results were considered satisfactory.

Upward loads are bounded by other loading conditions for the racks. Also, overload cutoff devices are installed on the fuel handling cranes in order to preclude uplift damage to the racks or fuel. Thus the design basis for upward loads was found to be acceptable.

Seismic Analysis of the Pool Structure

An analysis of the reinforced concrete pool structure was conducted by the licensee, and it was found that each pool floor is adequate to withstand the effects of added loads due to the new racks under seismic loads. The prestressed concrete girders (previously described) were found to be unaffected by the worst-case new rack loads.

We find that with respect to structural and mechanical design the subject modification proposed by the licensee satisfies the applicable requirements of General Design Criteria 2, 4, 61, and 62 of 10 CFR, Part 50, Appendix A and is acceptable.

2.2 Material Considerations

Discussion

We have reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water. In addition, our review has included an evaluation of the Boral neutron absorber material used in the high density storage locations for environmental stability.

The proposed spent fuel storage racks are fabricated primarily of Type 304 stainless steel, which is used for all structural components, except for a special low friction material (which is 99% graphite) used as a foot pad between the module and the support pad. Boral plates, used as a neutron absorber, are an integral nonstructural part of the basic fuel storage tubes. The Boral plates are sandwiched between the inner and outer wall of the storage tubes, and are not subject to dislocation, deterioration, or removal. The compartments in the storage tubes containing the Boral are exposed to the spent fuel pool environment through small openings formed during fabrication in the top and the bottom of each tube assembly.

The Brunswick Units 1 and 2 spent fuel storage pools contain high purity water. The chlorides are specified to be less than 0.2 ppm, the pH specified to be in the range 6.0 to 7.5, and the conductivity specified to be less than 1 μS. The design temperature of the water in the pool is 150°F, maximum. At most times during normal operation, the spent fuel pool temperature would never reach this design level, reaching a calculated maximum of 145°F immediately after a refueling operation and dropping rapidly thereafter until the next scheduled refueling.

The Type 304 stainless steel rack modules have been welded and inspected by nondestructive examination in accordance with the applicable provisions of ASME Boiler and Pressure Vessel Code, Section IX.

The licensee will perform a materials compatibility monitoring program consisting of two types of specimens: the first are 8" x 8" coupons of Boral covered with stainless steel, and the second consisting of 6" x 6" samples of Boral without stainless steel cladding. The stainless clad coupons have two sides open to permit water access to assess if any galvanic attack may be occurring between the aluminum and the stainless steel. It is particularly important to evaluate if this potential degradation mechanism might lead to loss of the neutron absorbing capabilities of the Boral. Sufficient coupons will be included to permit destructive examination of a sample on inspection intervals of 1 to 5 years over the life of the facility.

Evaluation

The Brunswick Units 1 and 2 spent fuel pools contain neutral, extremely high quality water in which all the materials of fabrication are expected to have good compatibility. The corrosion rate of Type 304 stainless steel in water of this quality and temperature is so low as to defy our ability to measure it. Galvanic effects between stainless steel, aluminum, and graphite are also unlikely in water of this quality as is stress corrosion cracking of weld sensitized stainless steel that may be present where the fuel storage cells are welded together. No instances of corrosion of these materials in water of this quality has been observed at any spent fuel storage pools in the country, some of which have been in operation for close to 20 years[1]. No loss of boron carbide from the Boral has been observed in material exposed to an environment similar to that in the Brunswick spent fuel storage pools for periods up to 20 years[2].

The venting of the cavities containing the Boral to the spent fuel pool environment will ensure that no gaseous buildup will occur in these cavities that might lead to distortion of the racks[2]. Flooding them will not introduce any significant corrosion problems where the aluminum is in contact with stainless steel in water of this quality. The low friction foot pads are made of a stable material, graphite, that will not be significantly affected by radiation or water and will not release significant quantities of potentially corrosive materials to the environment. The codes and standards used in fabricating and inspecting these new spent fuel storage racks should ensure their integrity and minimize the likelihood that any stress corrosion cracking of the racks themselves will occur during service.

The materials surveillance program spelled out by the licensee as outlined above will reveal any instances of corrosion of the Boral that might lead to loss of neutron absorbing power during the life of the new spent fuel racks. While we do not anticipate that such corrosion processes will occur, this monitoring program will ensure that, in the unlikely situation that corrosion should develop, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

From our evaluation as discussed above, we find that the corrosion that will occur in the Brunswick Steam Electric Station Units 1 and 2 spent fuel storage

pools will be of little significance during the remaining life of the plant. Components of the spent fuel storage pool are constructed of alloys which are known to have a low differential galvanic potential between them, and that have performed well in spent fuel storage pools at other boiling water reactor sites where the water chemistry is maintained to comparable standards to those in force at Brunswick. The proposed Materials Surveillance Program is adequate to provide warning in the unlikely event that deterioration of the neutron absorbing properties of the Boral will develop during the design life of the racks. Therefore, with the selection of the materials and water chemistry, we believe that no significant corrosion should occur in the spent fuel storage racks at Brunswick Units 1 and 2 for a period well in excess of the 40 years design life of the unit.

2.3 Installation and Heavy Load Handling Consideration

The results of the generic review of "Control of Heavy Loads at Nuclear Power Plants" (NUREG-0612) will not be completed until after the spent fuel storage modification has commenced. Thus, our review and evaluation of the heavy load handling operation has been limited to those activities associated with the spent fuel storage modification.

The installation of the spent fuel racks will be accomplished with the reactor building crane. The main hook of this crane is rated at 125 tons, while the heaviest load identified with the modification is the 15 x 17 storage rack weighing approximately 16 tons. Based on modifications to the crane and "upgrade" commitments by the licensee, the Staff previously concluded that the integrated design of the crane and controls with respect to the single failure criterion was acceptable. Based on the above, we conclude that the reactor building crane is acceptable for use for the modification.

A single failure proof lifting device for the handling of the new storage racks has been provided by the manufacturer. The lifting device has a rating of 39,000 lbs. and has been tested to 125% of capacity, while the heaviest storage rack is approximately 32,000 lbs. The removal of the existing storage racks will use the existing lifting device which has a safety factor of 5 on yield stress for design load of 9400 lbs. (maximum empty rack weight). The lifting apparatus such as slings, shackles and fittings are sized to maintain a minimum safety factor of 5 (based on ultimate strength - static load only). We conclude that the lifting devices and other apparatus used for the handling of the storage racks are adequate, and therefore, acceptable.

Specific load path instructions precluding the travel of heavy loads over stored spent fuel will be developed and implemented prior to the modifications. Also, handling procedures will be such that the storage racks which contain fuel will not be immediately adjacent to the rack being moved. We conclude that the proposed procedures are adequate, and therefore, acceptable.

Regarding operator training, qualifications and conduct, the licensee has stated that all operators are trained in accordance with the requirements of ANSI/B30.2-1976. The crane inspection, testing and maintenance program is also in conformance with the above referenced industry standard. We conclude that these are acceptable.

2.4 Spent Fuel Pool Cooling Considerations

System Description

Each BSEP Unit has an independent spent fuel pool and spent fuel pool cooling system (SFPCS). The major components of the SFPCS consists of two pumps in parallel, with one heat exchanger in series with each pump. These heat exchangers are cooled by the reactor building closed cooling water system. The heat removal capability of the SFPCS is 6.53×10^6 Btu/hr at 125°F and 12.0×10^6 Btu/hr at 150°F. The residual heat removal (RHR) system can be crosstied with the SFPCS in the event supplemental heat removal capability is required.

The refueling cycle for BSEP is an annual quarter core discharge of 140 fuel assemblies. Each assembly is assumed to have experienced a continuous power level of 4.35 MW_t prior to discharge. For both the normal refueling and full core discharge, the fuel will be subject to a 24-hour decay period after shutdown prior to its transfer to the spent fuel pool.

The licensee's calculated spent fuel discharge heat load to the pool, which was determined in accordance with the Branch Technical Position ASB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling," indicates that the expected maximum normal heat load following the last refueling is 14.1×10^6 Btu/hr. This heat load results in a maximum bulk pool temperature of 145.2°F. The expected maximum abnormal heat load following a full core discharge after the last normal refueling discharge is 29.2×10^6 Btu/hr. This abnormal heat load results in a maximum bulk pool temperature of 124.6°F if the RHR system supplements the SFPCS. A maximum pool temperature of 197.2°F is expected if only the spent fuel pool cooling system is used.

Evaluation

The American National Standard ANS 57.2 "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" indicates that the maximum pool temperature should not exceed 150°F under normal operating conditions with all storage full. Thus the RHR system will be operational and crosstied with the SFPCS prior to the discharge of a full core inventory into the pool.

The design of the storage pool is such that the fuel will always be covered with water. The top of the stored fuel is at an elevation lower than the bottom of the pool gate which separates the reactor well from the storage pool. Also, all piping which enter the storage pool are equipped with check valves and syphon breakers above the pool elevation to prevent inadvertent drainage. Normal makeup to the spent fuel pool is the demineralized water system via the SFPCS. In the event of a loss of the SFPCS, the heatup time to commence boiling is approximately 13 hours. This is sufficient time to provide 28 gpm emergency makeup to the pool by the fire system.

We have reviewed the calculated heat values and conclude that the heat loads are consistent with the Branch Technical Position ASB 9-2. The spent fuel pool cooling system performance and the available makeup systems have been reviewed and found to be acceptable.

2.5 Criticality Considerations

The spent fuel storage modules consist of a checkerboard array of double-walled stainless steel cans which are held together by stainless steel angles which are welded over the entire length of the cans. The space between the two walls of the can contains a neutron poison in the form of Boral sheets having a minimum areal concentration of 0.013 grams per square centimeter of boron-10. The spacing between storage locations is 6.563 inches. Calculations were performed with fuel having an infinite multiplication value of 1.35 in the standard core configuration at a temperature of 20°C (68°F). This is equivalent to a standard 8 x 8 fuel assembly having a uniform enrichment of about 3.2 weight percent U-235. This is a higher value than is expected for the reload fuels at their point of maximum reactivity.

The analyses are performed with the MERIT program - a three-dimensional Monte-Carlo code that uses the ENDF/B-IV cross-section set. This code has been benchmarked against a wide range of critical assemblies including BWR fuel criticals and criticals with flat plate boron absorbers. Comparison of calculation and experiment show that MERIT underpredicts the multiplication factor by about 0.5 percent with a standard deviation (1 σ) of 0.2 percent. The calculated effective multiplication factor for the nominal rack design is 0.867 ± 0.009 including the MERIT calculational bias and uncertainty.

Sensitivity calculations were performed in order to assess the effect of stainless steel thickness variations, temperature variations, center-to-center pitch, and locations of the assembly in the storage tube. The results of these studies showed that either the nominal case had the largest reactivity or that the uncertainty due to the variation was within the statistical uncertainty of the Monte-Carlo calculation. The effect of dropping a fuel assembly and a loss of fuel pool cooling have been investigated. For the former event the multiplication factor remains below 0.90 and for the latter case the 20°C water temperature case represents the maximum reactivity.

The interaction of the new storage racks with the existing system was investigated. The minimum separation between the new BWR storage racks and the existing PWR racks will be six inches. At this distance there is no significant neutron communication between the two systems. The maximum multiplication factor for the combined system is therefore that of the existing PWR racks - 0.95.

Based on our review we find that the reactivity aspects of the proposed BWR rack design are acceptable. Our bases were as follows:

1. The calculations were performed with a state-of-the-art code which was benchmarked by comparisons with critical experiments,

2. Conservative assumptions were made with respect to the input parameters to the code,

3. The effect of biases and uncertainties have been treated,

4. Interactions with storage racks existing in the pool have been treated,

5. The resultant calculation meets our acceptance criterion of less than or equal to 0.95 for the multiplication factor, and

6. Sufficient margin exists between the calculated multiplication factor and the acceptance criterion to account for any uncertainty due to the variation in the way the infinite multiplication factor of the stored assemblies is obtained (for example, a uniform enrichment design as opposed to a poisoned design partially burned up but having the same infinite multiplication factor as the uniform enrichment design).

We find that BWR fuel having an infinite multiplication factor less than or equal to 1.35 (at its most reactive point in life) in standard core geometry at 20° degrees Centigrade may be safely stored in the proposed racks. The 1.35 value shall be calculated for the axial segment of the assembly having the highest reactivity.

The licensee proposes to verify the presence of the Boral in the racks by scanning each storage location with a neutron source and detector. This is an acceptable procedure for verifying the presence of the Boral in the racks.

2.6 Spent Fuel Pool Water Cleanup Considerations

Description

The spent fuel pool cleanup system is incorporated as a part of the spent fuel pool cooling system. The spent fuel cooling system for each plant consists of two pumps, two heat exchangers, two filter demineralizers, two skimmer surge tanks, associated piping, valves and instrumentation. The skimmer surge tanks are designed to remove debris from the pool water and provide pump suction. The filter demineralizers (mixed bed resin) are designed to remove corrosion products, fission products, and impurities from the pool water. The demineralizers, like the pumps, are connected in parallel for operational flexibility. Pool water purity is monitored by a continuous conductivity meter installed on the inlet to the fuel pool demineralizers, and by periodic grab samples for laboratory analysis. Once a week, samples are taken for chemical and radiochemical analysis. Demineralizer resin will be replaced when either: (1) effluent conductivity equals influent conductivity at values above 1 μmho/cm, (2) effluent conductivity exceeds 1 μmho/cm by a significant margin, or (3) differential pressure reaches 25 psi.

The licensee indicated that no change or equipment addition to the spent fuel pool cleanup system is necessary to maintain pool water quality for the increase in fuel storage capacity.

Evaluation

Past experience showed that the greatest increase in radioactivity and impurities in spent fuel pool water occurs during refueling and spent fuel handling. The refueling frequency and the amount of core to be replaced for each fuel cycle, and frequency of operating the spent fuel pool cleanup system are not expected to increase as a result of high density fuel storage. The chemical and radionuclide composition of the spent fuel pool water is not expected to change as a result of the proposed high density fuel storage.

Past experience also shows that no significant leakage of fission products from spent fuel stored in pools occurs after the fuel has cooled for several months. To maintain water quality, the licensee has established the frequency of chemical and radiochemical analysis that will be performed to monitor the water quality and the need for spent fuel pool cleanup system demineralizer resin and filter replacement. In addition, the licensee has also set the chemical and radiochemical guidelines to be used in monitoring the spent fuel pool water quality and initiating corrective action. These guidelines are consistent with the reactor coolant Technical Specification water quality requirements.

The facility 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 November 1973. 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.

On the basis of the above, we determined that the proposed expansion of the spent fuel pool will not appreciably effect the capability and capacity of the spent fuel pool cleanup system. More frequent replacements of filters and demineralizer resin, if necessary, could offset any potential increase in the pool water as a result of the expansion of stored spent fuel. Thus we have determined that the existing fuel pool cleanup system with the proposed high density fuel storage (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool and thus meets the requirements of GDC 61 in Appendix A of 10 CFR Part 50 as it relates to appropriate systems to fuel storage; (2) is capable of reducing occupational exposures to radiation by removing radioactive products from the pool water, and thus meet the requirements of Section 20.1(c) of 10 CFR Part 20, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water within the filters and demineralizers, and thus meets Regulatory Position C.2.f(2) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the sources; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud from fluids through physical action. Therefore, no change to the spent fuel pool cleanup system is required.

2.7 Occupational Radiation Exposure

We have reviewed the licensee's plan for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for this operation is estimated by the licensee to be approximately 81 man-rem. This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job was being performed. In several instances he is conservative in his estimation of dose-rate and man-hours to perform a specific operation. Crud may be released to the pool water because of fuel movements during the proposed SFP modification. This could increase radiation levels in the vicinity of the pool and decrease the

clarity of the water. There will be a number of fuel movements in each pool during the modification. The plants have not experienced significant releases of crud to the pool water during refuelings when the spent fuel is first moved into the pools and the addition of crud to the pool water, from the fuel assembly and from the introduction of primary coolant water to the pool water, is the greatest. The licensee does not expect to have significant releases of crud to the pool water during the modification of the pools. The purification system for each pool, which has kept radiation levels in the vicinity of the pool to low levels, includes a filter to remove crud and will be operating during the modification of the pools. The pool floor will be vacuumed during the modification to remove particles which fall to the floor. The staff finds that the SFP modification can be performed in a manner that will ensure as low as is reasonably achievable (ALARA) exposures to occupational workers.

The licensee has proposed decontaminating the spent fuel racks which are removed and storing them on-site. The dose estimated for this activity is 6 man-rem.

We have estimated that the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies at both units on the basis of information supplied by the licensee and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at both units. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable levels and within the limits of 10 CFR Part 20. Thus, we find that storing additional fuel in the two pools will not result in any significant increase in doses received by occupational workers.

3.0 CONCLUSION

We have performed an evaluation of the licensee's proposed modifications based primarily on information provided to us in the licensee's basic supporting document. This document has been revised and supplemented during the course of our review in response to staff questions, and from meetings and discussions with the licensee, and to address new or more refined information regarding the proposed modification.

Our evaluation concludes that the proposed modification of the Brunswick Station Units 1 and 2 spent fuel storage is acceptable because:

(1) The structural and mechanical design for the proposed modification satisfies the applicable requirements of General Design Criteria 2, 4, 61, and 62 of 10 CFR Part 50, Appendix A and is acceptable.

(2) The compatibility of the materials and coolant used in the spent fuel storage pool is adequate based on tests, data, and actual service experience in operating reactors. The selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components and Criterion 62, by preventing criticality by maintaining structural integrity of components.

(3) The installation of the proposed fuel handling racks can be accomplished safely.

(4) The likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool is sufficiently small that no additional restrictions on load movement are necessary while our generic review of the issues is underway.

(5) The cooling system for each of the spent fuel pools has acceptable cooling capacity for normal fuel off loading. The spent fuel cooling system, as supplemented by the RHR system has acceptable cooling capacity for discharge of full core inventory into the pool.

(6) The physical design of the new storage racks will preclude criticality for any credible moderating condition.

(7) The existing SFP cleanup system is adequate for the proposed modification.

(8) The conclusions of the evaluation of the waste treatment systems are unchanged by the modification of the spent fuel pool.

(9) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the spent fuel pool would be negligible.

We conclude, then, 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 proposed license amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:	R. A. Hermann	B. Turovlin
	S. MacKay	P. Wu
	T. Cain	T. Chan
	R. J. Serbu	J. S. Boegli
	W. Brooks	O. Rothberg

Dated: December 15, 1983

References

1. J. R. Weeks, "Corrosion of Materials in Spent Fuel Storage Pools," BNL-NUREG-23021, July 1977.

2. J. R. Weeks, "Corrosion Considerations in the Use of Boral in Spent Fuel Storage Pools," BNL-NUREG-25582, January 1979.

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO.61TO FACILITY OPERATING LICENSE NO. DPR-71 AND SUPPORTING AMENDMENT NO.87TO FACILITY OPERATING LICENSE NO. DPR-62 RELATING TO THE MODIFICATION OF THE SPENT FUEL STORAGE POOL CAROLINA POWER & LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION AND DISCUSSION

The combined spent fuel storage capacity of the two nuclear units at Brunswick Station was originally 1440 BWR fuel assemblies, or storage for approximately 1.3 cores from each of the two units. This capability was later increased by a modular rack design to a maximum of 616 PWR or 1386 BWR assemblies. In actuality the installed storage capabity was for 304 PWR and 2088 BWR assemblies at the plant. This limited storage capability was in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a period of 3 to 5 years and then shipped offsite for reprocessing and recycling of the fuel.

Reprocessing of spent fuel did not develop as had been anticipated, however, and in September, 1975, the Nuclear Regulatory Commission (NRC, the Commission) directed the NRC staff (the staff) to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement would consider alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August, 1979. In the FGEIS, consistent with the long range policy, the storage of spent fuel is considered to be interim storage, to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of onsite fuel storage capacity by modification of the existing spent fuel pools. Applications for fifty such spent fuel capacity increases have been reviewed and approved. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage pool designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommends that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing spent fuel pools, other spent fuel storage alternatives are discussed

in detail in the FGEIS. The fnding of the FGEIS is that the environmental impact costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of the various alternatives reflect the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, that of shutting down the reactor when the spent fuel storage capacity is filled, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Impact Appraisal (EIA) addresses the environmental concerns related only to expansion of the Brunswick Station spent fuel storage pools. Additional discussion of the alternatives to increasing the storage capacity of existing spent fuel pools is contained in the FGEIS.

1.1 Description of the Proposed Action

By letter dated April 16, 1981, and supplemented by letters dated June 22 and November 23, 1981, March 16, April 5, May 20 and September 16, 1982 and February 23, March 31, and May 5, 1983, Carolina Power & Light Company proposed an amendment that would allow an increase in the licensed storage capacity of the two spent fuel pools to 3946 fuel assemblies. The storage capacity would be increased by replacing some existing racks with new, more compact, neutron absorbing racks. This would provide storage for spent fuel generated at Brunswick while maintaining full core off load capability at each unit through 1987.

The environmental impacts of the Brunswick facility, as designed, were considered in the NRC's Final Environmental Statement (FES) issued January 1974 relative to the continuation of construction and operation of the facility. The licensee later increased the storage capacity to a maximum of 616 PWR or 1386 BWR bundles. The environmental impact of this action was considered in an environmental impact appraisal issued with our authorization for this action in Amendment Nos. 8 and 30 issued August 26, 1977.

In this EIA we have evaluated any additional environmental impacts which are attributable to the currently proposed increase in the SFP storage capacity for the Station.

1.2 Need for Increased Storage Capacity

Spent fuel storage pools are provided for each of the two nuclear generating units at the Brunswick facility. The facility now has 304 PWR and 1908 BWR spaces provided by racks already installed and usable. Of the 1908 BWR spaces, 1080 spaces are occupied by spent fuel and 828 spaces are empty. For the Unit 2 maintenance outage now scheduled for Spring 1984, the full core of 560 assemblies needs to be removed and stored temporarily in order to safely and with minimum personnel exposure perform needed inspections and modifications. Of the 828 empty spaces available, 530 are located in the racks now installed in Unit 2. Obviously this number of empty spaces will not accommodate the full Unit 2 core. Further, a spent fuel cask is not available (not licensed) to move spent fuel from Unit 2 to Unit 1 and, therefore, no way is available to make use of the space available in the other unit. Therefore,

additional space is needed in the immediate future if Unit 2 is to be able to off load the full core to perform the required maintenance.

1.3 Fuel Reprocessing History

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shutdown in 1972 for alterations and expansion; in September 1977, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The pool is on land owned by the State of New York. NSF's lease with the State of New York expired in 1980 and their license has been suspended. The State of New York has requested the utilities who own the spent fuel presently stored in the pool to remove it. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate. The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois is licensed to store spent fuel. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris Operation was renewed for another 20 years; GE is not accepting any additional spent fuel for storage at this facility.

2.0 THE FACILITY

The principal features of the spent fuel storage and handling at Brunswick Station as they relate to this action are described here as an aid in following the evaluations in subsequent sections of this environmental impact appraisal.

2.1 The Spent Fuel Pool (SFP)

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core; also, they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

2.2 SFP Cooling System

The spent fuel and cooling system (SFPCS) for each unit at the Brunswick Station consists of two pumps in parallel, with a pump and heat exchanger in series. The heat removal design capability is 6.53×10^6 Btu/hr at 125°F and 12.0×10^6 Btu/hr at 150°F. The residual heat removal system (RHR) can be crosstied with the SFPCS in the event supplemental heat removal capability is required.

Heat is transferred from the spent fuel pool cooling system to the reactor building closed cooling water system. The reactor building closed cooling water system, in turn, transfers heat to the service water system. The RHR system is also a closed system cooled by service water. The service water system is a once-through cooling system in which well water or strained water from the Atlantic Ocean is supplied from pumps in the intake structure and returned to the ocean after removing heat from a number of systems, including the reactor building closed cooling water and the RHR systems.

2.3 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the NRC's Final Environmental Statement (FES) dated January, 1974. There will be no change in the waste treatment systems described in Section III.D.2 of the FES because of the proposed modification.

2.4 Spent Fuel Pool Cleanup System

The SFP cleanup system is part of the pool cooling system. It consists of a demineralizer with inlet and outlet filters, and the required piping, valves, and instrumentation. 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.

3.0 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

3.1 Nonradiological Consequences of the Proposed Action

The nonradiological environmental impacts of Brunswick, as designed, were considered in the FES issued January, 1974. Increasing the number of assemblies stored in the existing fuel pools will not cause any new nonradiological environmental impacts not previously considered. The amounts of waste heat emitted by each of the units as a result of the proposed increased spent fuel storage capacity will increase slightly (less than one percent), but will result in no measurable increase in impacts upon the environment.

3.2 Radiological Consequences of the Proposed Action

3.2.1 Introduction

The potential offsite radiological environmental impact associated with the expansion of spent fuel storage capacity at Brunswick has been evaluated.

During the storage of the spent fuel under water, both volatile and non-volatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54, which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominantly non-volatile at the temperature conditions that exist in pool storage. The primary impact of such non-volatile radioactive nuclides is their contribution of radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of the radionuclides in the pool water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent pool during refueling operations), or crud dislodged from the surface of the spent fuel during transfer from reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably.

A few weeks after refueling, the spent fuel cools in the pool so that the fuel cladding temperature is relatively cool, approximately 180°F. This substantial temperature reduction reduces the rate of release of fission products from the fuel pellets, and decreases the gas pressure in the gap between pellets and cladding, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on operational reports submitted by licensees, and discussions with storage facility operators, there

has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

3.2.2 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional fuel assemblies for a longer period of time would be the noble gas radionuclide Krypton-85 (Kr-85). As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no longer a significant release of fission products, including Kr-85, from stored fuel containing cladding defects. One hundred forty (140) fuel assemblies are expected to be stored following each March refueling at Unit 1 and each November refueling at Unit 2. Since space must be reserved to accommodate a complete reactor core unloading operation (nominally 560 fuel assemblies), and module spaces are reserved for PWR fuel assemblies, the useful pool capacity is 901 fuel assemblies at Unit 1 and 1243 fuel assemblies at Unit 2. At an input of 140 fuel assemblies per year, the storage capacity is approximately 9 years at Unit 2 and 6.5 years at Unit 1.

For the simplest case, we assumed that all of the Kr-85 that is going to leak from defected fuel is going to do so in the 12 month interval between refuelings. In other words, all of the Kr-85 available for release is assumed to come out of the fuel before the next batch of fuel enters the pool. Our calculations show that the expected release of Kr-85 from a 140 fuel assembly refueling is approximately 62 Ci each 12 months. As far as potential dose to offsite populations is concerned, this is actually the worst case, since each refueling would generate a new batch of Kr-85 to be released. As more and more fuel is added to the pool, one might think that this would increase the releases, but according to the terms of our model, this is not the case since all of the Kr-85 available for release has already left the defected fuel previously stored in the pool before the next batch enters, with the result that the annual releases are not cumulative but remain approximately the same. In other words, the enlarged capacity of the pool has no effect on the total amount of Kr-85 released to the atmosphere each year. Thus, we conclude that the proposed modifications will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above the 150 F used in the design

analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FES. Most airborne releases of tritium and iodine result from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than the spent fuel pool. Therefore, even if there were a higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increased stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If it is desired to reduce levels of radioiodine, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. In addition, the station radiological effluent Technical Specifications which are not being changed by this action, limit the total releases of gaseous activity.

3.2.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when the reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity in the SFP water to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two resin beds a year due to the increased operation of the spent fuel pool cleanup system. The annual average volume, per unit, of solid wastes shipped from the Brunswick Plant during 1978 through 1980 was 15,000 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 160 cubic feet per unit per year, the increase in total waste volume shipped would be approximately 1% and would not have any significant additional environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification are contaminated and will be disposed of as low level solid waste. We have estimated that approximately 7000 cubic feet of solid radwaste will be removed from the plant because of the proposed modification. Averaged over the lifetime of the plant this would increase the total waste volume shipped from the facility by less than 3%. This will not have any significant additional environmental impact.

3.2.4 Radioactive Material Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. Since the SFP cooling and cleanup system operates as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of this modification. The SFP demineralizer resin removes soluble radioactive material from the SFP water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins. If any radioactive material is transferred from the spent resin to the sluice water, it will be removed by the liquid radwaste system for processing. After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

3.2.5 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for the entire operation in estimated by the licensee to be about 81 man-rem. We consider this to be a reasonable estimate because it is based on realistic dose rates and occupancy factors for individuals performing a specific job during the pool modification. This operation is expected to be a small fraction of the total annual man-rem burden from occupational exposure.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel pool area from radionuclide concentrations in the SFP water and from the spent fuel assemblies.

The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Consequently, the occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less that one percent to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

3.3 Environmental Impact of Spent Fuel Handling Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated spent fuel handling accident, and a fuel shipping cask drop accident, in the SFP area, from those values previously reported in the Brunswick FES, based on the following considerations.

The heaviest identified load with this modification is a 15 x 17 rack weighing 16 tons, whereas the main hoist on the reactor building crane is rated at 125 tons. From a previous review we had concluded that the overhead crane load handling system and Technical Specifications meet our requirements and are acceptable. Spent fuel casks are of course not permitted over spent fuel stored in the pool. Further the licensee's spent fuel storage cask is not

currently licensed for use. The only items transported over spent fuel are other fuel assemblies, pool canal gates, and a fuel channel measuring device, none of which approach this weight capacity of 125 tons. We have concluded then that the likelihood of a heavy load handling accident is sufficiently small that the proposed modifications are acceptable, and no additional restrictions on load handling operations in the vicinity of the SFP are required.

3.4 Radiological Impacts to the Population

The proposed increase of the storage capacity of the SFP will not create any significant additional radiological effects to the population. The additional total body dose that might be received by an individual at the site boundary, and by the estimated population within a 50-mile radius, is less than 0.10 mrem/yr and 0.001 man-rem/yr, respectively. These doses are small compared to the fluctuations in the annual dose this population receives from background radiation. The population dose represents an increase of less than 0.01 percent of the dose previously evaluated in the FES for the Brunswick Steam Electric Plant, Units 1 and 2. We find this to be an insignificant increase in dose to the population resulting from the proposed action.

4.0 SUMMARY

The findings contained in the Final Generic Environmental Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, (the FGEIS) issued by the NRC in August, 1979, were that the environmental impact of interim storage of spent fuel was negligible, and the cost of the various alternatives reflect the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in spent fuel pool designs, the FGEIS recommended licensing spent fuel pool expansions on a case-by-case basis. Expansion of the spent fuel storage capacity at Brunswick Station does not significantly change the radiological impact evaluated by the NRC in the FES issued in January, 1974. As discussed in Section 3.4 of this EIA, the additional total body dose that might be received by an individual at the site boundary or the estimated population within a 50-mile radius is less than 0.10 mrem/yr and 0.001 man-rem/yr respectively, and is less than the natural fluctuations in the dose this population would receive from background radiation. The occupational exposure for the modifications (including rack decontamination for on-site storage) of the SFPs is estimated by the licensee to be 87 man-rem. This is conservative. Operation of the plant with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the total annual occupational exposure at the two units.

5.0 BASIS AND CONCLUSION FOR NOT PREPARING AN ENVIRONMENTAL IMPACT STATEMENT

We have reviewed the proposed modifications relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6. We have determined, based on this assessment, that the proposed license amendments will not significantly affect the quality of the human environment. Therefore, the Commission has determined that an environmental impact statement need not be prepared and that, pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Principal Contributors:

R. A. Hermann
T. Cain
R. J. Serbu
J. S. Boegli
P. Wu

Dated: December 15, 1983

U. S. NUCLEAR REGULATORY COMMISSION

CAROLINA POWER & LIGHT COMPANY

DOCKET NOS. 50-~~254~~ 324 AND 50-~~265~~ 325

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSE

AND NEGATIVE DECLARATION

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 61 and 87 to Facility Operating License Nos. DPR-71 and DPR-62, issued to Carolina Power & Light Company (the licensee), which revised the Technical Specifications for Operation of the Brunswick Steam Electric Plant, Units 1 and 2 (the facility) located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendments authorize changes to the Technical Specifications to allow an increase in the spent fuel storage capacity to a maximum of 3946 assemblies.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments in connection with this action was published in the FEDERAL REGISTER on May 27, 1981 (46 FR 28531). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Impact Appraisal related to the action and has concluded that an environmental impact statement is not

warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated January ;1974.

For further details with respect to the action see (1) the application for amendment dated April 16, 1981, as supplemented June 22 and November 29, 1981, March 16, April 5, May 20, and September 16, 1982, February 23, March 31, May 5 and September 20, 1983, (2) Amendment Nos. 61 and 87 to License Nos. DPR-71 and DPR-64, (3) The Commission's Safety Evaluation, and (4) The Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the Southport-Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 15th day of December, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing