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Dockets Nos. 50-269, 50-270
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Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 90, 90, and 87 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your request dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, November 3 and December 12, 1980.

These amendments allow an increase in the spent fuel storage capacity from 750 to a maximum of 1312 fuel assemblies in the Unit 1/2 common spent fuel pool through the use of neutron absorbing spent fuel racks.

Your June 24, 1980 letter stated that an additional cooling train is anticipated to be added to the Unit 1/2 spent fuel pool cooling system by April 1981; however, in no event will more than 342 spent fuel assemblies be stored in the subject pool until the additional cooling train is operable or unless prior approval is granted by the NRC.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and Notiee of Issuance/Negative Declaration are also enclosed.

Sincerely,

Original signed by P.B. Erickson

for Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 90 to DPR-38
2. Amendment No. 90 to DPR-47
3. Amendment No. 87 to DPR-55
4. Safety Evaluation
5. Environmental Impact Appraisal
6. Notice/Negative Declaration

B. ? E.K. See comment on EIA AE.

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

December 24, 1980

Docket No. 50-269/270 & 287

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: OCONEE UNITS 1, 2 AND 3

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendments Nos. 90, 90 & 87
Referenced documents have been provided PDR

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE	ORB#4:DL					
SURNAME	RIngram/cb					
DATE	12/30/80					

NRC FORM 102 (1-76)

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

December 24, 1980

Dockets Nos. 50-269, 50-270
and 50-287

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 90, 90 , and 87 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your request dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, November 3 and December 12, 1980.

These amendments allow an increase in the spent fuel storage capacity from 750 to a maximum of 1312 fuel assemblies in the Unit 1/2 common spent fuel pool through the use of neutron absorbing spent fuel racks.

Your June 24, 1980 letter stated that an additional cooling train is anticipated to be added to the Unit 1/2 spent fuel pool cooling system by April 1981; however, in no event will more than 342 spent fuel assemblies be stored in the subject pool until the additional cooling train is operable or unless prior approval is granted by the NRC.

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

Sincerely,

Peter B Erickson for
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 90 to DPR-38
2. Amendment No. 90 to DPR-47
3. Amendment No. 87 to DPR-55
4. Safety Evaluation
5. Environmental Impact Appraisal
6. Notice/Negative Declaration

cc w/enclosures: See next page

Duke Power Company

cc w/enclosure(s):

Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. Francis Jape
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

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Washington, D. C. 20036

cc w/enclosure(s) & incoming dtd.:
7/1, 7/25, 8/7, 8/14, 10/15, 10/31 & 11/3/80
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
License No. DPR-38

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

   A. The application for amendment by the Duke Power Company (the licensee) dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, and November 3 and December 12, 1980, 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 and paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 90 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8101100448

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 24, 1980

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
License No. DPR-47

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

   A. The application for amendment by the Duke Power Company (the licensee) dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, and November 3 and December 12, 1980, 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 and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 90 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

T M Novak

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 24, 1980

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-55

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

   A. The application for amendment by the Duke Power Company (the licensee) dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, and November 3 and December 12, 1980, 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 and paragraph 3.B of Facility Operating License No. DPR- 55 is hereby amended to read as follows:

   3.B 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

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 24, 1980

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 90 TO DPR-38

AMENDMENT NO. 90 TO DPR-47

AMENDMENT NO. 87 TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.8-2	3.8-2
5.4-1	5.4-1
5.4-2	5.4-2

Changes on the revised pages are indicated by marginal lines.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.10 The reactor building purge system, including the radiation monitor, RIA-45, which initiates purge isolation, shall be tested and verified to be operable immediately prior to refueling operations.

3.8.11 Irradiated fuel shall not be moved from the reactor until the unit has been subcritical for at least 72 hours.

3.8.12 Two trains of spent fuel pool ventilation shall be operable with the following exceptions:

a. With one train of spent fuel pool ventilation inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the operable spent fuel pool ventilation train is in operation and discharging through the Reactor Building purge filters.

b. With no spent fuel pool ventilation filter operable, suspend all operations involving movement of fuel within the storage pool or crane operations with loads over the storage pool until at least one train of spent fuel pool ventilation is restored to operable status.

3.8.13 a. Prior to spent fuel cask movement in the Unit 1 and 2 spent fuel pool, spent fuel stored in the first 36 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 55 days.

b. Prior to spent fuel cask movement in the Unit 3 spent fuel pool, spent fuel stored in the first 20 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 43 days.

3.8.14 No suspended loads of more than 3000 lb_m shall be transported over spent fuel stored in either spent fuel pool.

3.8.15 a. No fuel which has an enrichment greater than 3.5 weight percent U^{235} (46 grams of U^{235} per axial centimeter of fuel assembly) will be stored in the spent fuel pool for Unit 3.

b. No fuel which has an enrichment greater than 4.3 weight percent U^{235} (57 grams of U^{235} per axial centimeter of fuel assembly) will be stored in the spent fuel pool for Units 1 and 2.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation.

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Specification

5.4.1 New Fuel Storage

5.4.1.1 New fuel will normally be stored in the spent fuel pool serving the respective unit.

In the spent fuel pool serving Units 1 and 2, the fuel assemblies are stored in racks in parallel rows, having a nominal center-to-center distance of 10.65 inches in both directions. This spacing is sufficient to maintain a $K_{eff} \leq 0.95$ when flooded with unborated water, based on fuel with an enrichment of 4.3 weight percent U^{235}.

In the spent fuel pool serving Unit 3, the fuel assemblies are stored in racks consisting of stainless steel cavities which maintain a minimum edge-to-edge spacing of 3.95 inches between adjacent fuel assemblies. The neutron poisoning effect of the storage cavity material combined with the minimum 3.95 inches edge-to-edge spacing between adjacent fuel assemblies is sufficient to maintain a $K_{eff} \leq 0.95$ when flooded with unborated water based on fuel with an enrichment of 3.5 weight percent U^{235} or the equivalent.

5.4.1.2 New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center-to-center distance of 2' 1-3/4". One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.

5.4.1.3 New fuel may also be stored in shipping containers.

5.4.1.4 New fuel of enrichment not exceeding 2.9 weight percent U^{235} or the equivalent may be placed in dry storage in Unit 3 fuel storage racks in a checkerboard pattern, with fuel assemblies occupying only diagonally adjacent storage locations. Unused storage locations in a fuel storage module shall be covered by inserting a metal plate in the lead-in to prevent incorrect placement of fuel assemblies. This configuration is sufficient to assure a $K_{eff} \leq 0.9$ at all times.

5.4.2 Spent Fuel Storage

5.4.2.1 Irradiated fuel assemblies will be stored, prior to offsite shipment, in a stainless steel lined spent fuel pool.

The spent fuel pool serving Units 1 and 2 is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program.

Provisions are made in the Unit 1, 2 spent fuel pool to accommodate up to 1312 fuel assemblies and in the Unit 3 spent fuel pool up to 474 fuel assemblies.

5.4.2.2 Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

5.4.3 Except as provided in Specification 5.4.1.4, whenever there is fuel in the pool, the spent fuel pool is filled with water borated to the concentration that is used in the reactor cavity and fuel transfer canal during refueling operations.

5.4.4 The spent fuel pool and fuel transfer canal racks are designed for an earthquake force of 0.1g ground motion.

REFERENCES

FSAR, Section 9.7

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

INTRODUCTION

By letter dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, November 3, and December , 1980, Duke Power Company (DPC or the licensee) requested an amendment to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. The request would revise the provisions in the Station's common Technical Specifications (TSs) to allow an increase in Units Nos. 1 and 2 common spent fuel pool (SFP) storage capacity from 750 to a maximum of 1312 fuel assemblies through the use of neutron absorbing "poison" spent fuel storage racks.

The expanded storage capacity would allow the Oconee units to operate until about 1986 while still maintaining the capability for a full core discharge.

The major safety considerations associated with the proposed expansion of the SFP storage capacity for the Oconee Station are addressed below. A separate Environmental Impact Appraisal has been prepared as part of this licensing action.

DISCUSSION AND EVALUATION

Criticality Considerations

The licensee has provided an analysis of the criticality of the proposed storage racks. The analysis was performed by DPC with the KENO-IV code - a three dimensional Monte-Carlo program designed for reactivity calculations. Cross-section input to the code is from the ENDF/B-IV compilation which is processed by the AMPX system of codes. This analysis procedure has been verified by using it to calculate a series of 27 critical experiments. These experiments spanned the enrichment range of interest to the Oconee racks and included experiments with separated fuel assemblies having stainless steel and boral absorbers interposed. From this comparison a calculational bias and variability were determined.

In addition to the base case calculation, the effect of mechanical uncertainties on biases and uncertainties was examined. These included the pile-up of mechanical

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tolerances, particle self-shielding in the boron, the effect of bowing in the cans, etc. The calculational uncertainty and mechanical uncertainties were summed to obtain a total uncertainty. The result of the analysis is an effective multiplication factor for the racks of 0.9475 with all uncertainties included.

The effect of accidents on the reactivity of the racks has been analyzed. Storage of an assembly in a location other than analyzed is precluded by rack design. The effect of other accidents is dominated by the presence of a large boron concentration in the water so that the value of the effective multiplication factor is smaller for the accident configurations than the design value.

Conclusion on Criticality

We have reviewed the submittal and conclude that the rack design is acceptable from the criticality point of view. This conclusion is based on the following:

1. The analysis methods as used by DPC are state-of-the-art and have been verified by comparison with critical experiments which incorporate the main features of the rack design.

2. The uncertainties evaluated encompass those expected to be encountered. For some effects, the limiting conservative value has been used in the analyses. For others, sensitivity studies have been used to obtain an uncertainty in the rack multiplication factor.

3. Credible accidents have been considered and shown to have acceptable consequences.

4. The value of the effective multiplication factor meets our acceptance criterion, less than or equal to 0.95, when all uncertainties have been added.

Thus we conclude that any number of fuel assemblies of Babcock & Wilcox (B&W) 15 X 15 design can be stored in the racks provided that the uranium in the fuel has an enrichment no larger than 4.3 weight percent U-235.

Spent Fuel Cooling

The licensed thermal power for Oconee Units Nos. 1 and 2 is 2568 MWt each. DPC plans to refuel these reactors every 18 months at which times about 70 of the 177 fuel assemblies in the cores will be replaced. To calculate the maximum heat loads in the SFP, DPC assumed a 168-hour time interval between reactor shutdown and the time when either the 70 fuel assemblies in the normal refueling or the 177 fuel assemblies in the full core offload are placed in the SFP. For this cooling time, DPC used the method given in NRC Branch Technical Position APCSB 9-2 (BTP) to calculate maximum heat loads of 21.9×10^6 BTU/hr for a normal refueling and 34×10^6 BTU/hr for a full core offload.

The spent fuel cooling system presently consists of two pumps and two heat exchangers. Each pump is designed to pump 1000 gpm (5.0×10^5 lbs./hr.), and each heat exchanger is designed to transfer 7.75×10^6 BTU/hr from 125°F fuel pool water to 90°F Recirculating Cooling Water (RCW), which is flowing through the heat exchanger at a rate of 5.0×10^5 lbs./hr.

DPC states that this system will be sufficient to keep the SFP water temperature below 150°F, the pool design temperature, until April 1981, prior to the Oconee Unit 1 refueling in 1981, when an additional SFP cooling pump and heat exchanger of the same capacity will be installed. We find this acceptable.

Using the method given on pages 9.2.5-8 through 14 of the November 24, 1975, version of the NRC Standard Review Plan, with the uncertainty factor, k, equal to 0.1 for decay times longer than 10^7 seconds, we calculate that the maximum peak heat load during the refueling which would fill the pool could be 22×10^6 BTU/hr and that the maximum peak heat loads for a full core offload that essentially fills the pool could be 34×10^6 BTU/hr. This full core offload was assumed to be a fully irradiated core which was taken out of its reactor vessel 35 days after the other Oconee unit, which shares this SFP, had been refueled. We also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pool from 750 to 1312 is 1.9×10^6 BTU/hr. This is the difference in peak heat loads for the present and the modified pools.

We conclude that with the three pumps operating, as DPC has committed to provide by April 1981, the cooling system can maintain the fuel pool outlet water temperature below 125°F for the normal refueling offload that fills the pool and below 136°F for the full core offload that fills the pool. In the highly unlikely event that all three SFP cooling systems were to fail at the time when there was a peak heat load from a full core in the pool, we calculate that the maximum heatup rate of the SFP water would be 9.0°F/hr. Thus, if the water were initially at an average temperature of 125°F/hr, it would be more than nine hours before boiling would start. We also calculate that after boiling starts the required water makeup rate will be less than 70 gpm. We conclude that nine hours will be sufficient time to establish a 70 gpm makeup rate.

Conclusion on Spent Fuel Cooling

We conclude that the present two loop cooling system is adequate to handle the heat load of 342 spent fuel assemblies. The licensee has committed, in his June 24, 1980 letter, not to exceed this number of spent fuel assemblies in the Units 1 and 2 SFP until the additional cooling train is in service.

We conclude that the cooling capacity of the three loop system proposed by DPC for the Oconee Nuclear Station Units 1 and 2 SFP cooling system will be sufficient to handle the heat load that will be added by the proposed modifications. We also conclude that the incremental heat load due to this modification will not alter the safety considerations of spent fuel cooling from that which we previously reviewed and found to be acceptable.

Installation of Racks and Fuel Handling

In their July 1, 1980 proposal, DPC states that at the time of the installation of the new racks there will be 342 spent fuel assemblies in the pool. The licensee's installation plan is to both remove old racks and install new racks, under water, from the north end of the pool, making use of the existing Cask Storage Platform. The plan is to move the racks in the pool at an elevation which is lower than the top of any stored fuel assemblies, such that there will be no movement of racks over stored fuel.

Conclusion on Fuel and Rack Handling

We conclude that DPC's plan will insure that no racks will be moved over the spent fuel assemblies in the pool. After the racks are installed in the pool, the fuel handling procedures in and around the pool will be the same as those procedures that were in effect prior to the proposed modifications. On this basis, we conclude that the fuel and rack handling procedures are acceptable.

Structural and Seismic Loadings

The spent fuel storage rack is composed of individual storage cells made of stainless steel. Each cell has a lead-in opening which is symmetrical and is blended smooth. These racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. The cells within a module are interconnected to form an integral structure. Each rack module is provided with leveling pads which contact the SFP floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The SFP is constructed of reinforced concrete lined with stainless steel clad plate. No alteration was made to the pool design to accommodate more spent fuel. Rather, more fuel assemblies are fitted into the existing pool configuration by reducing spacing between the fuel assemblies and installing a neutron absorbing material.

The proposed modification for the spent fuel storage capacity expansion program has been reviewed in accordance with the NRC report "Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", issued April 1978 and revised January 18, 1979. The structural review consisted of an examination of the following areas: the proposed design criteria, the design loads and load combinations, methods of analysis, the dropped fuel accident, the material properties, the hydrodynamic effects, and the effect of increased loads on the floor slab and liner.

The spent fuel rack is made of stainless steel. The material properties for structural components of the spent fuel racks used in the analyses were taken from Section III of the ASME Code. Load combinations and acceptance limits are in conformance with the NRC Standard Review Plan, Section 3.8.4, and ASME Section III, Subsection NF.

The Oconee Units 1 and 2 SFP poison racks have been designed to meet the requirement for Seismic Category I Structures. The dynamic response of the fuel rack assembly during a seismic event produced the largest stresses among the loading conditions considered.

The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. The second phase is a response spectrum analysis of a detail rack assembly finite element model. The damping values used in the seismic analysis are two percent damping for an operating basis earthquake (OBE) and four percent damping for a safe shutdown earthquake (SSE) as specified in Regulatory Guide 1.61. The responses of the model from accelerations in three directions are combined by the SRSS method in the structural analysis. The loads in four major components (support pad assembly, botton grid, top grid, and fuel cell) are examined, and the maximum loaded section of each of these components is found. These maximum loads from the detail model are used in the structural analysis to obtain the stresses within the rack assembly.

The licensee has shown that, during a postulated earthquake, the fuel rack modules may slide laterally along the bottom of the pool. However, the magnitude of sliding was small enough so that the modules will not collide with the pool wall nor with each other. The calculation was performed using a non-linear code, WECAN. Certain aspects of the non-linear code, such as the sliding friction element, have not been fully reviewed by NRC. The licensee, therefore, supplied by letter dated December , 1980, a simplified analysis based on a linear response. The reanalysis showed the magnitude of sliding to be small when compared with gap spaces available between the rack modules and the pool wall. We find the gap spaces large enough to accommodate lateral module motion due to earthquake forces.

Two accident loading conditions are postulated for fuel handling crane uplift analysis. The first condition assumes that the uplift load is applied to a fuel cell. The second condition assumes that the load is applied to the top grid. Calculations show that for either condition, the resulting stresses are within acceptable stress limits. In order to ensure that the SFP liner will not be perforated, two accident conditions are evaluated. The first accident condition assumes that the weight of a fuel assembly, control rod assembly and handling mechanism (3,000 lbs.) impacts on the top of the rack. Calculations show that the impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cell funnels and the section of cell above the upper grid structure and the rack base plate/lower grid assembly. The second accident condition assumes that the fuel assembly falls straight through an empty cell and impacts the rack base plate from a drop height of 234 inches. The results of this analysis show that the impact energy is absorbed by the fuel assembly and the rack base plate. The SFP liner will not be perforated and the margin of safety is positive.

No alteration was made to the pool itself. The fuel pool concrete reinforcing steel, linear plate and welds connecting the inner plate to the fuel pool floor concrete embedments were analyzed based on consideration of the new racks and additional fuel. The results of the analysis were found to be acceptable and within the criteria given in the Final Safety Analysis Report (FSAR).

Conclusion on Structural and Seismic Loadings

The structural aspects of the spent fuel storage racks have been evaluated based upon NRC guidance provided in the report entitled, "Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," issued April 1978 and revised January 18, 1979. Based upon our review of the analyses and the design done by the licensee, we conclude that the rack structure itself, the supporting pool liner and slab, are capable of supporting the applied loads without exceeding relevant stresses of Subsection NF of ASME Section III or the FSAR design criteria. The proposed modifications to the Oconee spent fuel storage are in conformance with NRC requirements.

Fuel Cask Drop Accident Evaluation

We previously evaluated this accident in the June 1973 Safety Evaluation (SE) for the Oconee Nuclear Station for the original Units 1 and 2 SFP with 336 storage spaces, as described in the FSAR. In our SE dated June 19, 1979, related to the increase in capacity of this SFP from 336 to 750 storage spaces, we reevaluated the effects of a cask drop due to the closer spacing of spent fuel assemblies near the cask loading platform area. We concluded in our June 19, 1979 SE that the radiological consequences were mitigated by limiting the age of the spent fuel stored in the first 28 rows closest to the loading platform to a minimum decay time of 55 days.

To provide equivalent mitigation of such an accident for the SFP increase in capacity from 750 to 1312 storage spaces, the licensee has proposed to limit the age of fuel stored in the first 36 rows closest to the loading platform to a minimum decay time of 55 days. We find the 36-row limit equivalent to the previous 28-row limit. The 36-row limit and 55-day minimum decay interval are provided in proposed TS 3.8.13. We conclude that the consequences of a cask drop accident in the Units 1 and 2 SFP are not changed from those presented in our June 1973 and June 19, 1979 SEs with the implementation of the limits prescribed in TS 3.8.13 and are thus acceptable.

Materials Evaluation

The spent fuel racks in the proposed expansion will be constructed entirely of type 304 stainless steel, except for the nuclear poison material. The existing SFP liner is constructed of stainless steel. The high density spent fuel storage racks will utilize Boraflex sheets as a neutron absorber. Boraflex consists of 42 weight percent of boron carbide powder in a rubber-like silicone polymetric 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, the wrapper and the upper and lower spacer plates.

The upper end of the cell has a funnel shape flare for easy insertion of the fuel assembly. The wrapper surrounds the Boraflex material, but is open at the top and bottom to provide for venting of any gases that are generated. The Boraflex sheets sit in a square annular cavity formed by the square inner stainless steel tube and the outer wrapper. Each sheet is supported by lower spacer plate.

The pool contains oxygen-saturated demineralized water containing boric acid, controlled to a temperature below 150^{o}F.

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 not exceed a depth of 6.00 X 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 the 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, exposing Boraflex to 1.03 X 10^{n} 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. 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 corner 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 manufacturer's tests 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 weight percent of soluable 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 commited 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.

Conclusion on Materials

From our evaluation as discussed above we conclude that the corrosion that will occur in the Oconee SFP environment should be of little significance during the 40-year life of the plant. Components in the SFP are constructed of alloys which have a low differential galvanic potential between them and have a high resistance of general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation ard at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life of 40 years.

We further conclude that the environmental compatibility and stability of the materials used in the Oconee expanded SFP 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 SFP, 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. We therefore find that the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meet 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.

Occupational Radiation Exposure

We have reviewed the licensee's plan for the removal and disposal of the existing racks that were installed during a previous modification in 1979 and the installation of the new racks with respect to occupational radiation exposure. The occupational exposure for this operation is estimated by the licensee to be about 23 man-rems. We consider this to be a reasonable estimate because it is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification based on task comparisons with the previous re-racking. 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 is being performed. Although divers will be required during the modification, their expected cumulative dose equivalent will be about 10 man-rems.

The modification will be performed by arranging the spent fuel elements stored in the pool in such manner as to yield the lowest dose rates in the area to be occupied by divers while at the same time minimizing spent fuel movements or rearrangements of the assemblies which cause additional exposure to personnel performing and monitoring this operation. The existing spent fuel racks that will be removed from the SFP will be washed down and crated for disposal as low level radwaste at a licensed disposal site. The work to be performed will be performed in a manner consistent with "as low as is reasonably achievable" (ALARA) occupational exposures. All work will be performed in accordance with a radiation pre-plan to identify all protection requirements. Health physics personnel will be available to assure that ALARA radiation exposures prevail.

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 area from radionuclide concentrations in the SFP water and deposited on the SFP walls. 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. The occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible impact. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in additional exposure will not affect the licensee's ability to maintain individual occupational doses to ALARA and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

Radioactive Waste Treatment

The station 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 SE dated December 1970 for Oconee Unit 1 and in the SE dated July 1973 for Oconee Unit 2. There will be no change in the waste treatment systems or in the conclusions of the evaluations of these systems because of the proposed modification.

CONCLUSION ON SAFETY

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 that the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 24, 1980

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NO. 1, 2 AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

1.0 Introduction and Discussion

A Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575, Volumes 1-3) was issued by the Nuclear Regulatory Commission (NRC) August 1979. The NRC staff evaluated and analyzed alternatives handling and storage of spent light water power reactor fuel with emphasis on long range policy. Consistent with the long range policy, the storage of spent fuel addressed in the FGEIS 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 the onsite fuel storage capacity by modification of the existing spent fuel pools (SFPs). On the date of issuance of the FGEIS (August 1979), 40 applications for SFP capacity expansions were approved with the finding in each case that the environmental impact of the proposed increased storage was 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 SFPs, other spent fuel storage alternatives are discussed in detail in the FGEIS. The finding 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, where spent fuel generation is terminated, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Impact Appraisal (EIA) incorporates the appraisal of environmental concerns applicable to expansion of the Oconee Units 1 and 2 SFP. For additional discussion of the alternatives to increasing the storage capacity of existing SFPs, refer to the FGEIS. This EIA consists of three major parts plus a summary and conclusion. The three parts are: (1) descriptive material, (2) an appraisal of the environmental impacts of the proposed action, and (3) an appraisal of the environmental impact of postulated accidents.

1.1 Description of the Proposed Action

By application dated July 25, 1980, as supported by letter dated July 1, August 7 and 14, October 15 and 31, November 3 and December , 1980, Duke Power Company (the licensee) requested an amendment to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2 and 3. The proposed amendment would allow an increase in the storage capacity of the Oconee Units 1 and 2 common SFP from 750 to 1312 storage locations.

The environmental impacts of the ONS as designed, were considered in the Final Environmental Statement (FES) relative to the continuation of construction and operation of ONS issued March 1972. The purpose of this EIA is to determine and evaluate any additional environmental impacts which are attributable to the proposed increase in the SFP storage capacity of the Station.

1.2 Need for Increased Storage Capacity

The ONS consists of three generating units with a licensed power of 2,568 MWt for each unit. Units 1 and 2 share a common SFP with a storage capacity of 750 storage locations. Unit 3 has a pool with a capacity of 474 storage locations. All three units have 177 fuel assemblies in each core.

The modifications evaluated in this EIA are the proposals by the licensee to increase the pool storage capacity from 750 to 1312 spaces in the Oconee Units 1 and 2 common SFP.

The proposed increase would be accomplished by replacing the existing fuel storage racks with new, more compact, neutron absorbing racks. The proposed rack design uses a nominal 10.65-inch center-to-center spacing in each direction. The old racks had a nominal 13.75-inch center-to-center spacing in each direction. This modification would extend spent fuel storage capability past mid 1987 compared to early 1983 with the current capacity. The increase in capacity would extend the capability for a full core discharge from late 1982 to late 1986. This capability, while it is not needed to protect the health and safety of the public, is desirable in the event of a need for a reactor vessel inspection or repair. Such off-load capability would reduce occupational exposures to plant personnel.

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; on September 22, 1976, NFS informed the Commission that they were withdrawing from the nuclear fuel reprocessing business. 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, and the storage pool at West Valley, New York (on land owned by the State of New York and leased to NFS through 1980), are licensed to store spent fuel. The storage pool at West Valley is not full but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractural arrangements with NFS. GE is also not accepting any additional spent fuel for storage at the MO. Construction of the AGNS receiving and storage station has been completed. AGNS has applied for, but has not been granted, a license to receive and store irradiated fuel assemblies in the storage pool at Barnwell prior to a decision on the licensing action relating to the separation of facility.

1.3 Radioactive Wastes

The station 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 ONS FES dated March 1972. There will be no change in the waste treatment systems described in Section V.D and Appendix III.3 of the FES because of the proposed modification.

1.4 SFP Cleanup System

The SFP cooling and cleanup system consists of two circulation pumps, two heat exchangers, two filters, an ion exchanger, and the required piping, valves and instrumentation. This equipment is in two separate loops. The pumps draw water from the pool. This flow is passed through the heat exchangers and then returned to the pool. Approximately 100 gpm in each loop is bypassed through the filter and ion exchanger to maintain the clarity and purity of the water.

Therefore, because we expect only a small increase in radioactivity released to the pool water as a result of the proposed modification as discussed in Section 2.2, we conclude the SFP cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

2.0 Environmental Impacts of the Proposed Action

2.1 Nonradiological

The environmental impacts of ONS, as designed, were considered in the FES. Increasing the number of assemblies stored in the existing Units 1 and 2 common fuel pool will not cause any new environmental impacts. The amounts of waste heat emitted by ONS will increase slightly (less than one percent), resulting in no measurable increase in impacts upon the environment.

2.2 Radiological

2.2.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the Units 1 and 2 common spent fuel storage capacity was evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is the oldest fuel which has not been shipped and should have decayed at least six years. During the storage of the spent fuel under water, both volatile and nonvolatile 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 nonvolatile. The primary impact of such nonvolatile 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 radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the SFP purification system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the SFP so that the fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, 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 the operational reports submitted by the licensees or discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the MO (formerly Midwest Recovery Plant) at Morris, Illinois, or at the 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 SFP, 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.

2.2.2 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defective fuel. However, we have conservatively estimated that an additional 80 curies per year of Krypton-85 may be released when the modified pool is completely filled. This increase would result in an additional total body dose to an individual at the site boundary of less than .00008 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.003 man-rem/year. This is less than the natural fluctuations in the dose this population would receive from natural background radiation. Under our conservative assumptions, these exposures represent an increase of less than 0.05% of the exposures from the station evaluated in the FES for the individual at the site boundary and the population (Table VI.2). Thus, we conclude that the proposed modification will not have any significant nor measurable 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 of the Oconee Units.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above the 150°F during normal refuelings 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 modification from that previously evaluated in the FES. Most airborne releases from the station result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the SFP. Therefore, even if there were a higher evaporation rate from the SFP, the increase in tritium and iodine released from the station as a result of the increase in stored spent fuel would be small compared to the amount normally released from the station 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 from ONS including the releases from both pools.

2.2.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the filters and ion exchanger and be decay of short-lived isotopes. The activity is highest during refueling operations while reactor coolant water is introduced into the pool, and decreases as the pool water is processed through a filter and ion exchanger. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be an increase in solid radwaste from the SFP operations due to the modification, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by 51 cubic feet of resin per year from the ion exchanger (an additional resin bed per year) and the filters (two additional filters per year). The estimated annual average amount of solid waste shipped from the ONS from 1973 to 1977 was about 37,000 cubic feet per year. The annual average amount of solid waste shipped from Oconee Units 1 and 2 would be about 24,000 cubic feet per year. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification systems by about 51 cubic feet per year, the increase in total waste volume shipped would be less than 0.3% and would not have any significant environmental impact.

The present spent fuel racks to be removed from the SFP are contaminated and will be disposed of as low level waste. The licensee has estimated that less than 11,540 cubic feet of racks will be removed from the SFP because of the proposed modification. The old racks will be shipped, uncompacted, to the Barnwell site in South Carolina. The licensee is able to do this as the Barnwell facility does not have restrictions on compaction for in-state facilities such as ONS. This enables the licensee to avoid incurring a 5 man-rem dose that compaction would incur. The total waste shipped from the station will be increased by less than 0.5% per year when averaged over the lifetime of the station. This will not have a significant environmental impact.

2.2.4 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the station as a result of the proposed modification. The amount of radioactivity on the SFP filter and demineralizer might slightly increase due to the additional spent fuel in the pool, but this increase of radioactivity should not be released in liquid effluents from the station. The station radiological effluent Technical Specifications, which are not being changed by this action, restrict the total releases of liquid radioactivity from the station.

The cartridge filter removes insoluble radioactive matter from the SFP water. This is periodically removed to the waste disposal area in a shielded cask and placed in a shipping container. The insoluble matter will be retained on the filter or remain in the SFP water.

The demineralizer resins are periodically flushed with water to the spent resin storage tank. The water used to transfer the spent resin is decanted from the tank and returned to the liquid radwaste system for processing. The soluble radioactivity will be retained on the resins. If any activity should be transferred from the spent resin to this flush water, it would be removed by the liquid radwaste system.

Leakage from the SFP is collected in the leak collection system which consists of stainless steel channels imbedded in the concrete structure. The leakage is transferred to one of the waste storage tanks in the liquid radwaste system and is processed by the system before any water is discharged from the station. Before the waste storage tank, the leakage flows through an open basin where the flow could be observed. The basin is inspected periodically for signs of pool leakage. There have not been signs of leakage from the pool. Any leakage from the pool that could occur during the modification of the pool could also be detected through an increase in make-up water to the pool or an unusual increase in the level in a waste storage tank.

2.2.5 Occupational Exposures

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 licensee performed a similar pool modification in mid-1979 and bases the estimated exposures on observed exposures accumulated in the 1979 modification. The occupational exposure for the entire operation is estimated by the licensee to be about 23 man-rem. We consider this to be a reasonable estimate because it is based on observed dose rates and occupancy factors for individuals performing a specific job during the modification. This operation is expected to be a small fraction of the total 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 occupancy times and dose rates in the SFP area. 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. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than 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.

2.2.6 Impacts of Other Pool Modifications

As discussed above, the additional radiological environmental impacts in the vicinity of ONS resulting from the proposed modifications are very small fractions (less than 1%) of the impacts evaluated in the ONS FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that a SFP modification at any other facility should not significantly contribute to the environmental impact of the ONS and that the ONS SFP modification should not contribute significantly to the environmental impact of any other facility.

3.0 Environmental Impacts of Postulated 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 fuel handling accident or spent fuel cask drop accident in the SFP area from those values reported in the FES for Oconee Units 1 and 2 dated March 1972.

The environmental impact of a spent fuel shipping cask falling into the Oconee Units 1 and 2 SFP or Oconee 3 SFP is given in the EIA dated September 10, 1976. These impacts are not changed because of the proposed modification of the Oconee Units 1 and 2 SFP.

Additionally, the NRC staff has underway a generic review of load handling operations in the vicinity of SFPs to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because ONS will be required to prohibit loads greater than 3000 pounds (the normal weight of a fuel assembly, control rod and handling tool) to be transported over spent fuel in the SFP, we have concluded that the likelihood of any other heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is underway.

4.0 Summary

The FGEIS on Handling and Storage of Spent Light Water Power Reactor Fuel findings 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 SFP designs the FGEIS recommended licensing SFP expansions on a case-by-case basis. For ONS, expansion of the storage capacity of the SFP does not significantly change the radiological impact evaluated in the FES. As discussed in Section 2.2.2, the additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.00008 mrem/yr and 0.003 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 of the SFP is estimated by the licensee to be

22.7 man-rem. Operation of the station 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 station.

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.

Dated: December 24, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-269, 50-270 AND 50-287

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY

OPERATING LICENSES

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 90 , 90, and 87 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised Technical Specifications for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

The amendments allow an increase in the spent fuel storage capacity from 750 to a maximum of 1312 fuel assemblies in the Unit 1/2 common spent fuel pool through the use of neutron absorbing spent fuel racks.

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 Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on September 23, 1980 (45 FR 62948). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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The Commission has prepared an environmental impact appraisal for this action and has concluded that an environmental impact statement for this particular action is not warranted because it will not significantly affect the quality of the human environment.

For further details with respect to this action, see (1) the application for amendment dated July 25, 1980, as supplemented July 1, August 7 and 14, October 15 and 31, and November 3 and December 12, 1980, (2) Amendments Nos. 90 , 90, and 87 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, (3) the Commission's related 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., Washington, D. C. and at the Oconee County Library, 201 South Spring Street, Walhalla, South Carolina. 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 24th day of December 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter B Erickson

Peter B. Erickson, Acting Chief
Operating Reactors Branch #4
Division of Licensing