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September 29, 1983

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

Mr. H. B. Tucker
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 123 , 123, and 120 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station, Units Nos. 1, 2 and 3 (ONS-1, 2 & 3). These amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated March 10, 1983, supplemented by letters dated June 24 and 30, 1983, July 14, 1983, and August 8, 1983.

These amendments allow an increase in the spent fuel storage capacity from 474 to a maximum of 825 fuel assemblies in the Unit 3 spent fuel pool through the use of neutron absorbing (poison) spent fuel racks.

Your March 10, 1983 letter stated that an additional cooling train will be provided before the quantity of stored fuel assemblies exceeds the previously licensed capacity of 474 assemblies. The Commission finds this acceptable provided that the additional cooling train also be operational if more than 474 fuel assemblies are stored in the subject pool, unless prior approval is granted by the NRC to operate otherwise.

Copies of the Safety Evaluation and Environmental Impact Appraisal are also enclosed. Notice of Issuance/Negative Declaration will be included in the Commission's next Monthly Notice.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"
John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

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Enclosures:
1. Amendment No.123 to DPR-38
2. Amendment No.123 to DPR-47
3. Amendment No.120 to DPR-55
4. Safety Evaluation
5. Environmental Impact Appraisal

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DATE			9/21/83	9/21/83	9/27/83	9/23/83	9/27/83

Duke Power Company

cc w/enclosure(s):

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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. 123
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 March 10, 1983, as supplemented June 24 and 30, 1983, July 14, 1983, and August 8, 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 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. 123 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1983

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. 123
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 March 10, 1983, as supplemented June 24 and 30, 1983, July 14, 1983, and August 8, 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 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. 123 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1983

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. 120
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 March 10, 1983, as supplemented June 24 and 30, 1983, July 14, 1983, and August 8, 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 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. 120 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1983

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 123 TO DPR-38

AMENDMENT NO. 123 TO DPR-47

AMENDMENT NO. 120 TO DPR-55

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

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3.8-2	3.8-2
3.8-3	3.8-3
5.4-1	5.4-1
5.4-2	5.4-2

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.

c. This specification does not apply during reracking operations with no fuel in the spent fuel pool.

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 31 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 70 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 4.0 weight percent U^{235} (53 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 9.1.4 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.

Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above 1835 ppm. Although this concentration is sufficient to maintain the core $K_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The K_{eff} with all rods in the core and with refueling boron concentration is approximately 0.90. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.(3)

The off-site doses for the fuel handling accident are within the guidelines of 10 CFR 100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.8.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

<u>REFERENCES</u>

(1) FSAR, Section 9.1.4
(2) FSAR, Section 15.11.1
(3) FSAR, Section 15.11.2.1

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 $K_{eff} \leq 0.95$ 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 in parallel rows, having a nominal center-to-center distance of 10.60 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 4.0 weight percent U^{235}.

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.

5.4.2 Spent Fuel Storage

5.4.2.1 Irradiated fuel assemblies will be stored, prior to off-site 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 Units 1, 2 spent fuel pool to accommodate up to 1312 fuel assemblies and in the Unit 3 spent fuel pool up to 825 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 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 9.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 120 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

1.0 INTRODUCTION

By letter dated March 10, 1983, as supplemented June 24 and 30, 1983, July 14, 1983 and August 8, 1983, 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 the Unit No. 3 spent fuel pool (SFP) storage capacity from 474 to a maximum of 825 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 1990 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.

2.0 EVALUATION

2.1 CRITICALITY

2.1.1 EVALUATION

The Duke Power Company has provided an analysis of the criticality aspects of the proposed spent fuel pool expansion. The analysis was performed using the KENO-IV code, a Monte Carlo program optimized for reactivity calculations. The code has been benchmarked and verified with a large number of critical experiments which spanned the enrichment range of interest in Oconee. The cross sections used for the analysis were from the ENDF/B-IV. The moderator was assumed to be pure water at a density of 1 g/cm which would yield the largest reactivity within the temperature design limits of the pool. The comparison of the calculated and measured values yielded a bias (value of bias = 0, method uncertainty = .013 k) which is used in the calculated results. In addition, calculated uncertainties due to mechanical effects were examined. These include uncertainties due to mechanical design tolerances, particle self-shielding in the boron, bowing in the cans, etc. The uncertainties are for 95% probability at a 95% confidence level. The total uncertainty is the sum of its constituents (the square root of the sum of the squares). When the calculational bias and the sum of the uncertainties were included, the effective multiplication factor was found to be .9411.

The effects of accidents on the reactivity of the racks has been analyzed. Mislocation of an assembly is precluded by design. All configurations which could result from an accident are estimated to yield effective multiplication factors lower than the design value. The effective multiplication factor is dominated by the large amount of boron in the "boraflex" which is attached to each cell for which, however, no credit is taken but credit is taken for the boron dissolved in the pool water.

2.1.2 CONCLUSION

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

1. The Duke Power Company used analysis methods which are state-of-the-art and have been validated with critical experiments of arrangements incorporating the main design features of the racks,

2. An evaluated calculational bias (zero) and the sum of the expected uncertainties have been applied to the calculated value of the effective multiplication factor,

3. A series of credible accidents have been considered and were shown to have acceptable consequences, and

4. The value of the effective multiplication factor meets the acceptance criteria, i.e., less than or equal to .95 with the bias and the uncertainties taken into account.

We conclude that any number of fuel assemblies of the Babcock and Wilcox 15x15 design can be stored in the racks provided that the uranium in the fuel has an enrichment no greater than 4.00 w/o in U-235. We also conclude that the proposed revision to the Technical Specifications concerning the expansion of the storage capacity for the Oconee Unit 3 is acceptable.

2.2 MATERIALS

2.2.1 INTRODUCTION

The spent fuel racks in the proposed expansion would be constructed entirely of type 304 stainless steel, except for the nuclear poison material. The existing spent fuel pool 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 polymeric matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure. The major components of the assembly are the fuel assembly cells, the Boraflex material, 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°F.

2.2.2 EVALUATION

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 6.00×10^{-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.[2] 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 1.03×10^{11} rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to breakup of the Boraflex. Long term borated water soak tests at high temperatures were also conducted.[3] The tests show that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

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

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

2.2.3 CONCLUSION

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

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

We have reviewed the surveillance program and we conclude that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, provides reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. The materials surveillance program enacted by the licensee will reveal any instances of deterioration of the Boraflex that might lead to the loss of neutron absorbing power during the life of the new spent fuel racks. We do not anticipate that such deterioration will occur. This monitoring program will ensure that, in the unlikely situation that the Boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action. We therefore find that the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61 (having a capability to permit appropriate periodic inspection and testing of components) and Criterion 62 (preventing criticality by maintaining structural integrity of components and of the boron poison).

2.3 STORAGE RACKS

2.3.1 EVALUATION

The high density spent fuel storage racks are of the fixed poison, free standing modular design, and are designed to seismic Category I requirements. The two basic storage modules (8 x 10 and 8 x 12) weigh from 10 to 12 tons. Each module is composed of storage cells formed by two concentric stainless steel storage tubes with an inner core of Boraflex. The storage cells are arranged in a 10.60 inch center to center rectangular lattice configuration. The racks are also designed in such a manner that will not permit the insertion of fuel assemblies in other than prescribed locations. Further, they can withstand the maximum uplift forces produced by the fuel handling machine, and are designed such that the accidental dropping of a fuel assembly will not result in a geometry which can result in criticality.

2.3.2 CONCLUSION

We conclude that the proposed high density storage racks meet the requirements of GDC 2 and 62 with respect to seismic design considerations and prevention of criticality, and the guidelines of Regulatory Guides 1.13 and 1.29 with respect to fuel storage design and seismic design classification, and are, therefore, acceptable.

2.4 RACK HANDLING AND INSTALLATION

2.4.1 EVALUATION

The reracking modification will not commence until all fuel in the Unit 3 pool has been transferred to the Unit 1/2 pool, thus eliminating the concerns associated with carrying loads over stored spent fuel. The spent fuel pool expansion modification will utilize the 100 ton spent fuel pool crane (cask handling crane), a temporary construction crane, a lift bag and several special lifting devices. The handling of spent fuel racks constitutes the handling of heavy loads and thus is subject to the purview of NUREG-0612 - Control of Heavy Loads. The heaviest of the racks to be handled will be the existing 32 ton C-E double rack module. This module will be handled only by the temporary construction crane. The module will then be cut into individual rack assemblies of 16 tons each, prior to their removal from the pool. The Phase I review of the licensee's conformance to the guidelines of NUREG-0612 (whose analysis included the spent fuel pool crane) was completed and found acceptable as documented in our April 20, 1983 letter to the licensee. We conclude that the use of the 100 ton spent fuel pool crane is acceptable during the reracking modification. However, the temporary construction crane, the lift bag and various special lifting devices were not included in the original heavy loads analysis and thus are discussed below.

The temporary construction (T-C) crane is a gantry type bridge crane which traverses the length of the pool. The T-C crane has a design load rating of 40 tons to accommodate the heaviest of the existing modules (32 tons). The crane is designed to meet CMAA-70 and ANSI B30.2 criteria and will be load tested to 40 tons (e.g., 125% of the highest expected load). The hoist is reeved with stainless steel wire rope and employs a submersible block. We conclude that the use of the temporary construction crane is acceptable during the reracking modification.

The lift bag system is used to move the two interconnected modules located furthest from the cask storage pit to a point where they can be rerigged to the T-C crane. The lift bag system employs the lift bag itself, a compressor, an air regulator, and a globe valve. The lift bag is a pneumatic bladder which when inflated, will lift the rack assembly no higher than six inches off the pool floor. Adverse consequences are minimal in the event the lift bag fails in such a manner which results in a load drop. An analysis performed by the licensee for the NUREG-0612 evaluation determined that there was no equipment below the pool needed for safe shutdown. Further, the worst expected consequence would be the puncture of the pool liner rather than pool floor failure. We conclude that the use of the lift bag system is acceptable during the reracking modification.

The special lifting devices of concern are those associated with the existing C-E racks, and the new Westinghouse racks. The C-E lifting device has a load rating of 32 tons and will be load tested to 125% of rating capacity (i.e., 40 tons). The lifting device is built with a safety factor of 5 based on ultimate strength. The Westinghouse lifting device is the same as that which was used during the Unit 1/2 reracking. However, since the Unit 3 racks are lighter than those in the Unit 1/2 pool, using the same lifting device results in a safety factor of 5 based on yield strength. Lifting apparatus such as sling, shackles and fittings are sized as necessary to conform with the guidelines of NUREG-0612, Control of Heavy Loads. We conclude that the lifting devices and other apparatus used for the handling of the storage racks are adequate, and therefore, acceptable.

2.4.2 CONCLUSION

We conclude that the use of the proposed cranes and load lifting devices meet the requirements of GDC 4 and 61 with respect to protection of systems or components needed for safe shutdown from load drops, and the guidelines of NUREG-0612, Section 5.1.1, with respect to safe load handling practices.

Specific safe load paths for the spent fuel racks will be those already developed for cask handling. The height of the rack(s) over floor areas during the modification will be limited to six inches. Previous analysis by the licensee has shown that a load drop of a 25 ton spent fuel cask will not adversely affect the capability to safely shut down the plant. The implementation of safe load paths, operator training and qualification, and procedures will be consistent with NUREG-0612.

2.5 SPENT FUEL POOL COOLING SYSTEM

2.5.1 EVALUATION

The spent fuel pool cooling system (SFPCS) is designed to maintain the spent fuel pool water temperature, water inventory, clarity and chemistry at an acceptable level. It is designed to withstand the effects of a seismic event, and meets the requirements of Quality Group C classification. The existing system which is designed to accommodate the heat load from 474 fuel assemblies is composed of two trains with a heat removal capability of 15.5×10^6 Btu/hr at 125°F. The major components of the SFPCS consist of two pumps in parallel with one heat exchanger in series with each pump. These heat exchangers are cooled by the recirculating cooling water system. To supplement decay heat removal for the anticipated added heat load following the pool expansion modification, an additional cooling train with a heat removal capability of 7.75×10^6 Btu/hr at 125°F will be installed and operational prior to exceeding the current design inventory of 474 assemblies. The additional train is designed to meet the same seismic requirements and quality group classification as that of the original system.

The refueling cycle for Oconee 3 is an annual one-third core discharge of 59 fuel assemblies. Each assembly is assumed to have experienced a continuous power level of 14.5 MWt prior to discharge. For both the normal refueling and full core discharge, the fuel will be subject to a 72 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 normal refueling is 12.6×10^6 Btu/hr. This heat load results in a maximum bulk pool temperature of 140°F with two of three trains in operation (assumed single failure). The expected maximum abnormal heat load following a fuel core discharge after the last normal refueling discharge is 30.8×10^6 Btu/hr. This abnormal heat load results in a maximum bulk pool temperature of 150°F with all cooling trains in operation or 205°F with the loss of one train. The above maximum normal and abnormal heat load temperatures are within our guidelines. In the event of a loss of the SFPCS under maximum normal heat load calculations, the time to reach bulk pool boiling is approximately 15 hours and 5 hours respectively, which is sufficient time to provide emergency makeup to the pool. The required makeup of less than 70 gpm can be provided from alternate sources.

2.5.2 CONCLUSION

We have reviewed the calculated heat load 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. We conclude that the spent fuel pool cooling system meets the requirements of GDC 44 with respect to heat removal capability, and the guidelines of Regulatory Guide 1.13 with respect to system design considerations.

2.6 STRUCTURAL ASPECTS

2.6.1 INTRODUCTION

Oconee Unit 3 is a two-loop B&W PWR. The plant is founded on rock. The spent fuel pool serves Unit 3 exclusively and is located in the auxiliary building. The pool is a concrete box and is rectangular in plan view.

Inside dimensions are approximately 58 ft long by 24 ft wide x 42 ft deep (maximum dimensions).

The bottom of the pool is elevated above the basemat and the inside is at elevation 802 ft. The basemat is at elevation 758. In general the walls of the pool extend to the basemat. The north end of the pool (cask area) rests on a massive concrete pier (21.5 ft x 31 ft) which extends to basemat. The floor and the walls of the pool vary in thickness. The bottom of the pool is a minimum of 4.5 ft thick and the walls are a minimum of 3.5 ft thick. The pool is lined with a 1/4 inch thick stainless steel watertight liner plate. A leak-chase channel system is provided.

The existing fuel storage racks are to be replaced by ten new poisoned racks. These are free standing box-shaped structures with individual rectangular cells to store the fuel. The 8 cell by 10 cell rack is approximately 14 ft high by 9 ft wide by 7 ft long. The racks are constructed of type 304 stainless steel. The cells are constructed of cold formed 0.075 inch thick cold formed sheet. The cells are supported by and welded to a top and bottom grid of structural tubing. The bottom grid rests on and is welded to a heavy base plate which is in turn supported by and welded to four corner leveling pads.

2.6.2 EVALUATION

The racks were designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979 (referred to hereafter as the "NRC Position"). Section III, Division 1, Subsection NF of the ASME code was the basis of design of the racks.

The existing concrete pool structure was originally designed in accordance with ACI 318-63. The pool and liner were reevaluated in accordance with the original plant design criteria for the new rack loads.

Rack structural materials are in conformance with the requirements of the ASME code. Loads and load combinations for the design of the racks were found to be in accordance with the NRC Position. Loads and load combinations for the analysis of the pool structure were found to be in accordance with the original plant FSAR commitment and are acceptable.

The seismic loads are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. This was based on a 0.1g SSE and a 0.05g OBE. Damping values for the racks were taken as 2 percent for OBE and 4 percent of SSE. Impact effects due to fuel bundle/rack interaction as well as rack/pool floor interaction were included in the analysis.

A separate fuel assembly drop accident analysis was performed. A 3000 pound object was postulated to impact the top of the rack from a height of 6 feet. The same object was postulated to drop 234 inches through a cell and impact the bottom of the rack.

2.6.2.1 DESIGN AND ANALYSIS PROCEDURES

a. Racks

First a seismic time history analysis of a non-linear 2-dimensional model was conducted. The model consisted of spring, mass, damping friction, and gap elements to simulate a fuel bundle in a simplified model of a rack. The fuel assembly-to-cell impact loads, support pad lift-off values, rack sliding, and overall rack response were determined from the non-linear analysis. Coefficients of friction were varied between minimum and maximum possible values in order to determine worst case conditions of sliding and tipping respectively. Rack-to-rack impacts were precluded by spacing the racks beyond maximum possible excursion. In order to account for 3-dimensional effects, the results of independent orthogonal loadings were combined by the SRSS method.

Next, a seismic response spectrum analysis of a 3-dimensional finite element model of the racks, using inputs from the results of the non-linear analysis, and superimposed with other applicable loads, was conducted. Design stresses and safety margins for appropriate components in the racks were tabulated and found to be acceptable.

b. Pool

The spent fuel pool was reanalyzed for new rack loads. Results of key structural calculations comparing actual with allowable stresses for the analysis of the pool with the new rack loads with superimposed thermal and dead loads were provided. The factors-of-safety for the pool and the liner were found to be acceptable.

2.6.3 CONCLUSION

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

2.7 SPENT FUEL CASK MOVEMENT AND FUEL HANDLING ACCIDENTS

2.7.1 INTRODUCTION

In our Safety Evaluation (SE) dated September 1976, accidents involving the movement of the spent fuel cask were evaluated on the basis of the cask being handled in the spent fuel pool and with fuel present in the pool. This SE assumed 76 spent fuel assemblies could be damaged and that they had undergone a minimum aging time of 43 days. Since the issuance of the September 1976 SE, there have been Technical Specifications covering both the Unit 1/2 common spent fuel pool and also the Unit 3 spent fuel pool with regard to the number of rows of stored spent fuel potentially impacted by a cask movement accident and also the aging time of the fuel in these rows.

2.7.2 EVALUATION

The licensee has indicated that for this proposed reracking, the modifications to the pool will be accomplished with both the spent fuel cask and presently stored fuel removed from the pool altogether. The presently stored fuel will have been transferred to and stored in the Unit 1/2 common spent fuel pool prior to any work commencing on the presently existing racks. Thus, the possibility of potential accidents involving the spent fuel cask and stored fuel during the reracking has been precluded. With regard to the potential for an accident involving the stored fuel and the spent fuel cask once the reracking has been accomplished, the licensee has proposed increasing the present limit of 20 rows of fuel aged a minimum of 43 days to a new limit of 31 rows of fuel aged a minimum of 70 days. This requirement is incorporated in proposed TS 3.8.13. In the licensee's worst case accident analysis, a hoist cable failure would potentially cause the cask to be deflected into the pool wall and the yoke and load block could be deflected into the spent fuel. There are 128 cans under the projected cask, yoke, and block impact area. In considering the cans adjacent to the impact area, a total of 486 cans can potentially suffer a loss of integrity during a cask drop accident. The licensee's analysis for such an accident indicates that the worst radiological consequences experienced would result from 100% of the activity contained in the fission gases trapped in gaps in the fuel stored in the locations being released into the pool water. The exclusion area boundary dose, taking no credit for ventilation system filtration, would be 0.1 rem whole body and 55 rem to the thyroid. These doses are well below 10 CFR 100 limits and the licensee has demonstrated compliance with Standard Review Plan Section 15.7.4.

2.7.3 CONCLUSION

We conclude that the consequences of a cask drop accident and fuel handling accident in the Unit 3 spent fuel pool are not changed from those previously evaluated. With the implementation of the limits prescribed in TS 3.8.13, the staff finds the proposed reracking acceptable.

2.8 OCCUPATIONAL RADIATION EXPOSURE

2.8.1 INTRODUCTION

We have reviewed the licensee's plan for the removal and disposal of the low density racks, and 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 22 person-rems. This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification.

2.8.2 EVALUATION

Throughout the SFP modification operation, the licensee's personnel exposure controls will be administered in accordance with the licensee's health physics control procedures to assure as low as is reasonably achievable exposures (ALARA) to workers. The procedures include pre-job planning and worker briefings, checking water clarity, extensive surveys of the work areas, vacuuming the pool floor, walls and fuel rack surfaces around diver working areas and removing all spent fuel assemblies stored in the pool prior to diving operations. In addition, the licensee has developed specific operating procedures for divers to assure that their doses are ALARA.

The licensee has presented two alternative plans for the removal and disposal of the old racks. These are: (1) decontamination of the old racks prior to disposal as non-radioactive waste or (2) transfer of the old racks to an authorized burial site. The licensee will follow ALARA guidelines for workers regardless of which disposal method is chosen.

Based on the manner in which the licensee will perform the modification, and relevant experience from other operating reactors that have performed similar SFP modifications, we conclude that the Oconee Nuclear Station, Unit No. 3 SFP modification can be performed in a manner that will ensure ALARA exposure to workers.

We have estimated the increment in onsite occupational dose for normal operations after the pool modification which can result from the proposed increase in stored fuel assemblies at Unit 3. The spent fuel assemblies themselves contribute a neglible amount to dose rates in the pool area because of the depth of water shielding the fuel.

The proposed increase of the storage capacity of the SFP would 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 population within a 50-mile radius, is estimated to be less than 0.10 mrem/yr and 0.02 person-rem/yr, respectively. These doses are extremely 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 Oconee Nuclear Station, Unit No. 3. We find this to be an insignificant increase in dose to the population resulting from the proposed action.

Similarly, the proposed increase in storage capacity of the SFP would not affect radiological impact to the work force significantly. The dose to plant workers at Oconee, over the years 1974 to 1981, has averaged about 900 person-rems/year. The total projected worker dose is 22 person-rems, which is about one-fourteenth of the normal annual rate.

2.8.3 CONCLUSION

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 conclude that storing additional fuel in the pool will not result in any significant increase in doses received by workers.

3.0 CONCLUSIONS

Based on our review, we conclude that the proposed modified fuel storage design of the Oconee Unit No. 3 spent fuel pool meets our requirements. The proposed increase in the spent fuel pool storage capacity to a maximum of 825 fuel assemblies (maximum allowed enrichment of 4.0 weight percent U-235) through the use of neutron absorbing (poison) spent fuel racks meets the requirements of the General Design Criteria, as discussed above, of Appendix A to 10 CFR Part 50 and is, therefore, acceptable. Based on our review, we have also determined that the proposed TS changes for the Oconee Nuclear Station's Common TS's are acceptable.

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

Dated: September 29, 1983

The following NRC personnel have contributed to this Safety Evaluation:
L. Lois, B. Turovlin, T. Chan, O. Rothberg, M. Lamastra, J. Nehemias, J. Suermann.

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 120 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

1.0 INTRODUCTION AND DISCUSSION

The storage capacity of the Unit No. 3 spent fuel pool at the Oconee Nuclear Station (ONS) is 474 fuel assemblies. The original design capacity of the pool was 216 fuel assemblies and the increase from 216 to the presently authorized 474 fuel assemblies was approved on December 22, 1975 by means of an amendment to the Unit's operating license. The amendment also included a supporting Safety Evaluation (SE) and an Environmental Impact Appraisal (EIA). The limited storage capacity was in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a few years and then be 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 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, 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 Oconee Nuclear Station, Unit No. 3 spent fuel storage pool. 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 application dated March 10, 1983, and supplemented by letters dated June 24 and 30, July 14, and August 8, 1983, Duke Power Company proposed an amendment that would allow an increase in the licensed storage capacity of the Unit No. 3 spent fuel pool from 474 to 825 fuel assemblies. The storage capacity would be increased by replacing the existing racks with new, more compact, neutron absorbing racks, similar to those installed in the Unit 1/2 spent fuel pool in December 1980. This would provide storage for spent fuel generated at ONS while maintaining full core off load capability through 1990.

The environmental impacts of the ONS as designed, were considered in the Final Environmental Statement (FES) relative to the continuation of constuction 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 1312 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 those proposed by the licensee to increase the pool storage capacity from 474 to 825 spaces in the Oconee Unit 3 SFP.

The proposed increase would be accomplished by replacing the existing fuel storage racks with new racks as mentioned above. The proposed rack design uses a nominal 10.60-inch center-to-center spacing. The old racks had a nominal 14.09-inch center-to-center spacing. This modification would extend spent fuel storage capability to October 1991 compared to September 1988 with the current capacity. The increased capacity would extend the capability for a full core discharge from January 1988 to March 1990. 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.

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 plant is on land owned by the State of New York. NFS'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 the Oconee Nuclear 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 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.

2.3 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 3.2, 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.

3.0 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

3.1 NONRADIOLOGICAL

The environmental impacts of ONS, as designed, were considered in the FES. Increasing the number of assemblies stored in the existing Unit 3 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.

3.2 RADIOLOGICAL

3.2.1 INTRODUCTION

The potential offsite radiological environmental impact associated with the expansion of spent fuel storage capacity at Oconee Nuclear Station 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 appears to be radionuclides that were present in the reactor coolant system prior to refueling (which become mixed with water in the spent fuel 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.

For the simplest case, we assumed that all of the Kr-85 that is going to leak from defective 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. 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 defective 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.

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. 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. The licensee has chosen as a first alternative to decontaminate the racks and eventually sell the racks as scrap material. A second alternative, should the first not prove feasible, would be to dispose of the contaminated racks as was done for the 1980 rerack of the Unit 1/2 SFP. If the disposal of the present Unit 3 racks is required, it is estimated that the ten racks would comprise approximately 8,250 cubic feet (as compared to the last Unit 1/2 rerack of fourteen racks comprising about 11,540 cubic feet of waste) of waste to be disposed of as low level waste. The old racks would 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.

3.2.4 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.

3.2.5 OCCUPATIONAL RADIATION 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 occupational exposure for the entire operation is estimated by the licensee to be 22 person-rems. We consider this to be a reasonable estimate because it is based on dose rates and occupancy factors for individuals performing specific jobs during the modification and experience gained by the Oconee Nuclear Station, Units 1 and 2 Spent Fuel Pool Expansion. The dose due to this operation is expected to be a small fraction of the total annual person-rem estimated for operating the Station.

We have estimated the increment in onsite occupational doses which may result from the proposed increase in stored fuel assemblies. These are based on information supplied by the licensee and by the utilization of relevant assumptions for occupancy times and for dose rates in the spent fuel pool area. The spent fuel assemblies themselves contribute a negligible amount to the dose rates in the pool area because of the depth of water shielding the fuel. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modifications should add less than one precent to the total annual occupational radiation exposure and conclude that storing additional fuel in the pool will not result in any significant increase in doses received by workers.

3.2.6 RADIOLOGICAL IMPACTS TO THE POPULATION

The proposed increase of the storage capacity of the SFP would 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 estimated to be less than 0.10 mrem/yr and 0.02 person-rem/yr, respectively. These doses are extremely 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 Oconee Nuclear Station, Unit No. 3. We find this to be an insignificant increase in dose to the population resulting from the proposed action.

3.2.7 ENVIRONMENTAL IMPACT OF SPENT FUEL CASK/SPENT FUEL HANDLING ACCIDENTS

Prior to commencing any removal of racks for the proposed modification, the spent fuel in the Unit No. 3 SFP will have been transferred to the Unit 1/2 SFP for storage. Also, the spent fuel cask is stored away from the pool so that prior to and during the modifications the possibility of an accident involving the fuel assemblies from either the spent fuel cask or other loads is precluded. Once the modifications are complete even though the pool will accommodate a larger inventory of spent fuel, we have determined that the use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the March 1972 FES for the Oconee Nuclear Station.

The environmental impact of a spent fuel shipping cask falling into either of the Oconee SFPs (Unit 1/2 or Unit 3) is given in the EIA dated September 10, 1976. These impacts are not changed because of the proposed modification of the Oconee Unit No. 3 SFP.

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. Due to 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 Oconee Nuclear Station does not significantly change the radiological impact evaluated by the NRC in the FES issued in March 1972. As discussed in Section 3.2.6 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 0.02 person-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 SFP is estimated by the licensee to be 22 person-rems. 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 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 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: September 29, 1983