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November 16, 1989

Docket Nos.: 50-269, 50-270
and 50-287

Posted

Amdt. 177 to DPR-47

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 177, 177, AND 174 TO FACILITY OPERATING LICENSES DPR-38, DPR-47, and DPR-55 - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TACS 68026/68027/68028)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 177, 177, and 174 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Technical Specifications in response to your request dated March 31, 1988, as supplemented.

The Technical Specifications (TS) added by these amendments establish requirements for movement of a dry storage fuel transfer cask in Oconee Units 1, 2 and 3 spent fuel pools. In addition, the changes will allow storage of spent fuel at the Oconee Independent Spent Fuel Storage Installation (ISFSI) when licensed. Authorizations for the ISFSI required under the provisions of 10 CFR Part 72 are being handled by the Commission's Office of Nuclear Material Safety and Safeguards.

A copy of our Safety Evaluation and Notice of issuance are also enclosed.

Sincerely,
John B. Hopkins, Acting Project Director

for Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 177 to DPR-38
2. Amendment No. 177 to DPR-47
3. Amendment No. 174 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

OFFICIAL RECORD COPY
[OCONEE AMEND ISFSI]
*See previous concurrence

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| *LA:PDII-3 | *PM:PDII-3 | *DREP:PRPB | *DET:SPLB | *OGC | D:PDII-3 for |
| RIngram | LWiens:sa | JWigginton | CMcCracken | | DMatthews |
| 10/31/89 | 10/31/89 | 10/31/89 | 11/01/89 | 11/15/89 | 11/16/89 |

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated March 31, 1988, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 16, 1989

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

# DUKE POWER COMPANY

## DOCKET NO. 50-270

## OCONEE NUCLEAR STATION, UNIT 2

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated March 31, 1988, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 16, 1989

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

# AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated March 31, 1988, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 174 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 16, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 177

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 177

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

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

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 33 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 70 days.

c. Prior to dry storage transfer cask movement in the Unit 1 and 2 spent fuel pool, spent fuel stored in the first 64 rows of the pool closest to the cask handling area shall be decayed a minimum of 65 days.

d. Prior to dry storage transfer cask movement in the Unit 3 spent fuel pool, all spent fuel stored in that pool shall be decayed a minimum of 57 days.

3.8.14 No suspended loads of more than 3000 lbm 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 1950 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).

REFERENCES

(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.

5.4.2.3 Spent fuel may also be stored in Oconee Nuclear Station Independent Spent Fuel Storage Installation.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE DPR-47

AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

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

## 1.0 INTRODUCTION

The spent fuel pools (SFPs) at Oconee Units 1, 2 and 3 will not permit additional spent fuel storage beyond 1990. Therefore, the licensee, Duke Power Company, is constructing an Independent Spent Fuel Storage Installation (ISFSI) for separate onsite storage of spent fuel. The licensee intends the ISFSI to provide for interim onsite dry storage of Oconee spent fuel until the Department of Energy (DOE) provides for permanent offsite storage of such fuel. The licensee plans to use the NUHOMS-24P system (Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, HUM-002, Rev. 1, July 1988) for long-term dry storage. In this system, 24 spent fuel assemblies (SFAs) will be stored in a stainless steel dry shielded canister (DSC). The DSC is filled with helium and sealed after insertion of the SFAs; the helium serves as a medium for transferring the heat generated by the stored assemblies to the exterior of the DSC. The DSC is then stored in a concrete horizontal storage module (HSM) which is designed to hold the DSC for 20 years. The DSC may be removed from the HSM for transshipment to the DOE storage facility or to an intermediate facility for removal of SFAs which can then be transferred to the permanent storage facility. Duke Power plans to build ten HSMs, initially, with the option to build a total of eighty-eight (88). Eighty-eight HSMs are sufficient to store 2112 SFAs which will permit Oconee to continue operation to the end of its operating life without the need for any additional storage space. Nominally, the SFAs to be stored in the ISFSI must have decayed for a minimum period of 10 years and generate 0.66 kw of heat or less per SFA (the specifications for SFAs to be stored in the ISFSI are identified elsewhere). The licensee noted that these criteria were "...derived to ensure that the peak fuel rod temperatures, surface doses and subcriticality are below the design values."

Use of the ISFSI involves operations both within the fuel building and outside. The evaluation contained herein deals with those operations within the fuel building as part of the 10 CFR Part 50 license. The operations outside of the fuel building are reviewed separately, under the 10 CFR Part 72 license criteria. This amendment will allow storage of spent fuel at the Oconee ISFSI when licensed. Authorizations for the ISFSI required under the provisions of 10 CFR Part 72 are being handled by the Commission's Office of Nuclear Material Safety and Safeguards.

An abbreviated list of operations within the fuel building includes:

(1) The transfer cask is lifted from the trailer,
(2) The DSC is placed into the transfer cask cavity,*
(3) The cask and DSC are moved into the decon pit,
(4) The transfer cask with DSC is transferred to the cask pool,
(5) The DSC is loaded with 24 SFAs,
(6) The DSC is fitted with its top end shield plug,
(7) The transfer cask, together with DSC, is raised out of the pool to permit draining,
(8) The top end shield plug is welded to the DSC,
(9) The DSC is completely drained and vacuum dried,
(10) The DSC is filled with helium,
(11) Both the helium fill and water drain connectors are seal welded,
(12) The DSC top cover plate is welded over the top end shield plug,
(13) The transfer cask top cover is bolted on,
(14) The transfer cask/DSC assembly is transferred to the decon pit,
(15) The transfer cask/DSC assembly is transferred to the truck bay area,
(16) With the trailer in place and fuel building doors closed, the transfer cask/DSC assembly is lowered onto the trailer,
(17) With the transfer cask/DSC assembly properly positioned on the trailer, the fuel building doors are opened.

*The licensee has retained the option to move the transfer cask and DSC separately into the decon pit. In that case, operation (2) will be modified as follows:

(2) The transfer cask is transferred to the decon pit,
(2a) The DSC is transferred to a location above the transfer cask,
(3) The DSC is lowered into the cavity in the transfer cask.

The licensee has prepared a Safety Analysis Report and revised Technical Specifications (TS) to permit use of the new, 100-ton transfer cask and DSC. The revised TS were submitted by letter dated March 31, 1988, and amplifying information was provided by licensee letters dated February 7, March 21, April 11, June 22, June 27, July 17 and September 8, 1989.

The requested changes would require spent fuel to be decayed for a minimum of 65 days in order to occupy the first 64 rows of the pool closest to the cask handling area prior to dry storage transfer cask movement in the Unit 1 and 2 SFP. Similarly, the change pertaining to the Unit 3 SFP requires that all the spent fuel in that pool be decayed a minimum of 57 days prior to dry storage transfer cask movement. In addition, the proposed changes would allow storage of spent fuel in the ISFSI wehn licensed.

## 2.0 EVALUATION OF OPERATIONS

### 2.1.1 Maximum Fuel Cladding Temperatures

The licensee's contractor, Nutech, performed a thermal analysis of the transfer cask containing the DSC. Nutech used the Heating-6 computer program to determine the temperature of the transfer cask, DSC, SFAs and fuel elements within the SFA.

The NUHOMS-24P transfer cask (used to transport the DSC) and DSC shell were modeled as long, composite cylinders.

2.1.2 Determination of Maximum Fuel Cladding Temperatures

Nutech determined that the maximum fuel temperature expected during the process of loading the DSC with SFAs and drying it would occur during the process of evacuating the DSC to the design vacuum pressure of 3 torr. In this case, Nutech calculated a maximum fuel cladding temperature of 410°C. The maximum calculated fuel cladding temperature of 410°C was considered to be acceptable in comparison with the Nutech acceptance criterion for the dry cask assembly of 570°C for the short term (48 hours). Nutech calculated fuel cladding temperature for one further case, that for loss of the liquid neutron shield in the cask. Here, the maximum clad temperature was found to be 421°C, which was also acceptable.

The NRC staff's review of the contractor's calculations and storage temperature criteria for long-term maximum fuel temperature criteria is being conducted separately as part of the 10 CFR Part 72 license application. In order to assure maintenance of the long-term criterion, the licensee will inspect the transfer cask to assure the presence of the liquid neutron barrier before loading the DSC with SFAs and will check the DSC to assure that it is not evacuated when sealed.

2.2 Heavy Loads Concerns

2.2.1 Safe Load Paths

The licensee indicated that the transfer cask and DSC, handled together or separately, will not pass over any safety-related equipment or spent fuel when being moved from the transfer trailer to the SFP and back. The NRC staff finds the safe load paths to be acceptable.

2.2.2 Procedures and Training

The licensee provided an outline of the procedure to be used in transferring the DSC and transfer cask to the SFP, loading the DSC with SFAs, drying the DSC, filling and assembling the DSC, and transferring the transfer cask/DSC. The licensee will provide operator training and load handling instructions so as to assure reliable operation. Based on its review, the NRC staff finds that adequate procedures and training will be in place.

2.2.3 Special Lifting Devices

There are two special lifting devices used in conjunction with the transfer cask: the transfer cask lift beam and the crane hook lift adaptor. The lift beam is used to adapt the transfer cask to the 100 ton crane hook during upending and moving the transfer cask inside the fuel building. The lift adaptor is attached between the crane hook and transfer cask lift beam after the transfer cask is placed on the cask pit platform. The licensee reported that the adaptor is designed to permit lowering of the cask into the cask pit from the pit platform without wetting the 100 ton crane hook and block.

Both of these devices will be designed, built, and maintained in accordance with the criteria of ANSI 14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 1000 Pounds (4500 kg) or More for Nuclear Materials."

There is one device used solely with the DSC: the DSC lifting rig. The DSC lifting rig is used to upend the DSC on the trailer and lift it into the transfer cask. It is not used to transfer or move the DSC when loaded with SFAs, nor is it carried over safety-related equipment or spent fuel. Therefore, the DSC lifting rig need not be designed in accordance with the provisions of ANSI 14.6. The NRC staff finds the special lifting devices to be acceptable.

2.2.4 100 Ton Crane

The 100 ton crane has been previously reviewed and found acceptable by the NRC staff for use in carrying 100 ton loads. This review is documented in a Safety Evaluation dated April 20, 1983.

2.2.5 Load Handling Accidents

The licensee stated that the cask would not be carried over spent fuel. Nevertheless, the licensee evaluated, in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," the extent of potential accidents if the crane or handling equipment were to fail. This is discussed below.

2.2.5.1 Cask Drop in Oconee Spent Fuel Pools

The licensee analyzed a fall wherein the cask, yoke and yoke block are deflected into the Unit 1/2 SFP. In such a case, the licensee calculated that 1024 SFAs would be damaged (64 rows with 16 SFAs in a row) by assuming that the 220 fuel storage cells directly under the falling parts buckle and deflect into adjacent cells until the total energy of the falling cask is absorbed. Note that a total number of 1298 fuel assemblies can be stored in the Unit 1/2 pool. For a cask drop in the Unit 3 SFP, the licensee assumed that all 825 fuel cells would be damaged.

The NRC staff finds the number of fuel elements assumed damaged in a cask drop in either the Unit 1/2 SFP or the Unit 3 SFP to be conservative and, thus, the licensee's analysis is acceptable.

The radiological effects of a cask drop into either SFP are evaluated in Section 3.0, Radiological Accident Analysis.

2.2.5.2 Potential Criticality Caused by Cask Drop in Pools

The licensee analyzed the potential for criticality in the SFPs in the event of a transfer cask drop accident by using both the approach identified in NUREG-0612 and analyses to confirm the results obtained by use of NUREG-0612. The licensee used Tables 2.2-3 and 2.2-4 (entitled, "$K_{eff}$ for 3.5 w/o U-235 Fuel Under Different Accident Conditions and $K_{eff}$ for 5.0 w/o U-235 Fuel Under Different Accident Conditions") in NUREG-0612 to derive approximate $K_{eff}$ values for the

Unit 1/2 pool and Unit 3 pool. No credit was taken for pool boron concentration (minimum 1950 ppm), rack boroflex, and metal rack wall construction. It was also assumed that the racks and fuel had been crushed to maximize $K_{eff}$. The licensee subsequently took credit for the pool boron, boroflex and stainless steel walls to determine $K_{eff}$ under the assumed damage conditions in the confirmatory analyses, which used a specific neutronic analysis for each pool with the following assumptions:

1) An infinite array of fuel assemblies is crushed together into an optimum geometrical configuration.

2) Fuel assemblies affected are unirradiated with an enrichment equal to the maximum allowable for storage at Oconee.

3) A minimum technical specification fuel pool boron concentration.

The licensee made a series of calculations which covered cases of varied fuel pin pitch simulating crushing of the racks and SFAs. The calculated maximum $K_{eff}$ case for the Unit 1/2 pool was slightly greater than 0.96 while the calculated maximum $K_{eff}$ for the Unit 3 pool was found to be slightly less than 0.92.

These values are within NRC staff guidelines for ensuring post-accident (cask drop) subcriticality and are therefore acceptable.

2.2.5.3 Potential Criticality From Fall Onto Trailer

The possibility exists for a 40-50 foot drop of a transfer cask containing a fully loaded DSC onto the transfer trailer in such a manner as to rupture the cask. Damage to the DSC is also possible.

However, it is not possible for the 24 fuel elements in the DSC to be arranged in such a manner as to form a critical array while surrounded by helium or air. This is the case if the transfer cask were to be dropped while being loaded onto the trailer. The NRC staff, therefore, finds that nuclear criticality is not an issue from a transfer cask fall onto the transfer trailer.

2.2.5.4 Potential Criticality While Loading and Transporting the DSC

The licensee stated that DSC internals were designed to provide nuclear criticality safety during wet loading and unloading operations. The NRC staff noted that loading even unirradiated fuel into the DSC, together with optimum moderation would result in a $K_{eff}$ below 0.98 as long as borated water containing 1810 ppm was used to fill the DSC. Since the licensee will use water with a minimum boron content of 1950 ppm, nuclear criticality is prevented. It is also noted that the DSC will be drained of the water content within 50 hours of loading SFAs into the DSC. Therefore, the NRC staff finds that nuclear criticality is not an issue when transporting a DSC since no moderator is present.

2.2.5.5 Potential Damage to the SFP in the Event of a Transfer Cask Drop

The licensee stated that the SFP concrete floor slab is designed to withstand the 100 ton cask drop. The licensee indicated that the pool is founded on rock. The licensee stated that localized concrete could be crushed and the liner plate ruptured in the cask impact area. The licensee further stated that such localized deformation would not permit leakage beyond the damage area. However, for the purpose of calculation, the licensee assumed a gap of 1/64 inch for the perimeter of the ruptured liner plate (308 inches). Based on this area, the licensee calculated a loss of 21.3 gallons per day through the postulated separation. This value is within the capacity of SFP water makeup sources. The NRC staff finds the licensee's analysis of postulated fuel pool damage and water loss to be conservative and therefore acceptable.

2.3 Special Precautions

2.3.1 Fuel Clad Temperature

In order to avoid long-term fuel cladding temperatures in excess of 340°C (644°F), it is necessary that the following precautions be taken:

(1) The SFAs to be loaded into the DSC must meet the stated criteria for ISFSI storage,

(2) The DSC must be filled with helium, and

(3) The transfer cask liquid neutron shield must be intact.

The licensee will, therefore, qualify SFAs in accordance with specified criteria. The licensee stated that a list of the SFAs to be provided for a particular DSC will be sent to fuel handling personnel who will visually verify the identification numbers of the SFAs to be loaded. Verification will be performed by two different persons after which these SFAs will be removed from the SFP and installed in the proper DSC. The SFA identification will again be checked after the DSC has been completely loaded. A similar verification system will be used in case the SFAs from a previously loaded DSC are to be returned to the SFP. The licensee also stated that SFAs suspected of having fuel pins with cladding failure will be examined visually (by camera). Those showing gross cladding failure or structural damage will be excluded from storage in the ISFSI. The NRC staff finds this approach to ensuring proper SFA loading in the ISFSI to be acceptable.

In addition, the licensee agreed to insert a step in the procedure for loading and installing a DSC in an HSM which would require verification that the neutron shield water jacket is full. The NRC staff finds this to be acceptable.

3.0 RADIOLOGICAL ACCIDENT ANALYSIS

The licensee's accident analysis used assumptions that were, in part, bounding within the context of the requested TS amendments. The NRC staff used most of the same assumptions in an independent accident analysis. Specifically, for analytical purposes, a cask drop in the Units 1 and 2 SFP was assumed to damage all of the 1024 fuel storage cells in the first 64 rows. Two full fuel loads

(354 fuel assemblies) were assumed to have decayed one year. Likewise, in the Unit 3 SFP, one full core load (177 fuel assemblies) was assumed to have decayed 57 days and the remaining 648 fuel assemblies to have decayed one year. Should the SFP of Oconee Units 1 and 2 contain more than 354 fuel assemblies or the SFP of Oconee Unit 3 contain more than 177 fuel assemblies that have decayed less than one year, movement of the dry storage transfer cask in or about the affected area would exceed the scope of this Safety Evaluation. Accordingly, prior to movement of the dry storage transfer cask in the unanalyzed circumstances described above, the licensee is advised to seek NRC approval.

The staff's independent analysis assumed a gap, puff release with the characteristics described in Regulatory Guide 1.25, except no peaking factor was used since entire core loads were assumed to be damaged. Thus, the average power of the fuel assemblies was appropriate. The dispersion for a two-hour release at the exclusion radius was taken as 2.2E-04 sec/cubic meter. Since the accident was modeled as a puff release, the calculated dose after two hours is equal to the dose for the entire duration. The calculated doses from the staff's analysis for Oconee Units 1 and 2 were 0.25 rem to the whole body and 54 rem to the thyroid. For Oconee Unit 3, the results were 0.22 rem to the whole body and 54 rem to the thyroid. These calculated doses, taken separately, are well within (i.e., less than 25 percent as referenced in Regulatory Guide 1.25) of the 10 CFR Part 100 limits of 25 rem to the whole body and 300 rem to the thyroid. Overall, it is the staff's judgment and conclusion that, with respect to 10 CFR Part 100, the proposed changes to the TS are acceptable.

Since Kr-85 would be a major component of a gap release from decayed fuel, the staff also analyzed the skin doses from the postulated accidents. At the exclusion boundary, the skin dose was calculated to be 28 rem and 22 rem due to the postulated dry storage transfer cask drop accidents in the Oconee 1 and 2 and Oconee 3 SFPs, respectively. Skin doses in the spent fuel buildings and elsewhere onsite could be considerably greater. The skin dose from beta radiation is approximately a factor of 100 greater than the dose from gamma radiation where Kr-85 is the sole radionuclide released. The licensee's radiation control program to ensure adequate protection from these beta hazards is described below.

Oconee has a beta radiation protection training program in place which covers dosimetry of, and protection from, point, distributed, and cloud sources. This training covers the response to a fuel handling accident involving old fuel with a release of Kr-85. In the event of a spent fuel accident or a SFP radiation monitor alarm, personnel are trained to leave the SFP area at once and not to re-enter the area without appropriate approval. In addition, personnel involved with the handling of the ISFSI transfer cask will receive enhanced training which will emphasize the recognition of a Kr-85 release, the need for immediate evacuation, and conditions and precautions for re-entry. Oconee's procedures describe the necessary response to spent fuel accidents.

Oconee is equipped with both fixed and portable radiation monitors which are sensitive enough to detect a Kr-85 release from a spent fuel accident involving old fuel. Process monitors located just outside of the pool are sensitive enough to detect airborne Kr-85 causing a skin dose rate of 0.1 mrad/hr. The unit vent gas monitor for the fuel pool area is also beta sensitive. Both of these monitors are calibrated using beta emitters with energies bracketing the Kr-85 beta energy. The area radiation monitor located on the spent fuel bridge is primarily gamma sensitive, but would respond to large releases that could potentially occur with a cask drop event. Oconee also has several types of portable instruments available that can be used for Kr-85 skin dose rate measurements in response to a spent fuel accident involving old fuel. Finally, the thermoluminescent dosimeters (TLDs) used at Oconee will accurately measure skin doses due to Kr-85 exceeding 250 mrads if the appropriate calibration factor is used. On the basis of the above, the NRC staff finds that the licensee has an adequate training program and appropriate radiation monitoring instrumentation for detection of Kr-85 in the event of a fuel handling accident involving old fuel.

The staff also analyzed the radiological consequences of a gaseous release resulting from the drop of the dry storage transfer cask loaded with 24 fuel assemblies that were decayed for 7.5 years (shortest decay period used in the licensee's Safety Analysis Report). For this analysis, the staff assumed that the gap release of all 24 assemblies was from fuel assemblies with a 1.65 peaking factor, and 10 percent of the halogens and noble gases were released, except for Kr-85 and I-129. The release fraction for these latter two nuclides was assumed to be 30 percent. No decontamination factor from water was used since the storage cask is dry. The 7.5 year decay left only Kr-85 in sufficient quantity to yield a calculated dose from a puff release. The calculated doses at the exclusion area boundary were 0.005 rem to the whole body, 0.67 rem to the skin and 0.0 rem to the thyroid. These results are well within the limits of 10 CFR Part 100 and are acceptable in this respect.

## 4.0 OCCUPATIONAL EXPOSURE CONTROLS

The licensee's ALARA program follows the general guidelines of Regulatory Guides 1.8, 8.8, and 8.10. The ALARA design features used to design and build the ISFSI are the same as those used to design and build the Oconee nuclear plant and are described in Chapter 12 of the Oconee Final Safety Analysis Report (FSAR). The primary goal of these design features is to minimize exposure to radiation so that the total exposure to personnel is maintained as low as is reasonably achievable.

Some of the design considerations incorporated in the construction of the ISFSI include the use of thick concrete walls on the HSM to reduce the surface dose to below an average of 20 mr/hr, the use of a lead shield plug on the ends of the DSC to reduce the dose to workers performing drying, sealing, and loading operations, and the use of a shielded transfer cask for handling and transportation operations of loaded DSCs. The HSM has external shielding blocks over its air outlets to reduce direct and streaming doses, and the entire ISFSI is located well away from occupied areas. The NRC staff finds these design features acceptable for minimizing doses to plant personnel during the handling, transportation, and storage of spent fuel in the ISFSI.

The licensee will use portable shielding during DSC drying/welding operations to limit streaming from the top end shield plug/DSC annulus. The licensee has performed a detailed time-motion study to estimate the total personnel doses during the fuel handling and transfer activities associated with one NUHOMS-24P module (1 cask cycle). The expected cumulative dose is 1047 person-mrems (approximately one person-rem) per cask cycle. The licensee plans to process seven modules over the first three-year period that the ISFSI is in operation and four to five modules each year after the first three years. The estimated doses associated with use of the ISFSI are a small fraction of the approximately 1075 person-rems per year that the Oconee plant has averaged over the past five years for all three units. The NRC staff finds this estimate to be reasonable and acceptable.

## 5.0 STAFF FINDINGS

Based on its review, the NRC staff concludes that the steps taken by the licensee to move spent fuel into the dry storage cask, to insert the dry storage cask into the transfer cask, and to move the transfer cask/dry storage cask assembly in the spent fuel building comply with the guidelines of NUREG-0612 and applicable portions of the Standard Review Plan for such transfer and, therefore, are acceptable.

The proposed TS changes assure compliance with the guidelines of NUREG-0612 and the analysis assumptions for the postulated transfer cask drop into the Unit 1/2 or Unit 3 SFPs, respectively, and permit spent fuel storage in the ISFSI. The NRC staff finds these proposed changes consistent with the licensee's Part 50 safety analysis for the ISFSI and therefore acceptable for inclusion in the Oconee Units 1, 2 and 3 TS.

## 6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, the Commission has determined that the issuance of these amendments will have no significant impact on the environment (54 FR 43369).

## 7.0 CONCLUSION

The Commission issued a Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for Hearing, which was published in the Federal Register (53 FR 26122) on July 11, 1988, and consulted with the state of South Carolina. No requests for hearing were received, and the State of South Carolina did not have any comments.

We have concluded, based on the consideration 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 mannner, 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.

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Dated: November 16, 1989

# UNITED STATES NUCLEAR REGULATORY COMMISSION

## DUKE POWER COMPANY

## DOCKET NOS. 50-269, 50-270, AND 50-287

## NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 177, 177, and 174 to Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 issued to Duke Power Company (the licensee), which revised the Technical Specifications for operation of the Oconee Nuclear Station, Units 1, 2, and 3 (the facility) located in Oconee County, South Carolina. The amendments were effective as of the date of issuance.

The amendments revise the Technical Specifications to establish requirements for movement of a dry storage fuel transfer cask in Oconee Units 1, 2 and 3 spent fuel pools. In addition, the changes authorize storage of spent fuel at the Oconee Independent Spent Fuel Storage Installation (ISFSI). Authorizations for the ISFSI required under the provisions of 10 CFR Part 72 are being handled by the Commission's Office of Nuclear Material Safety and Safeguards.

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

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on July 11, 1988 (53 FR 26122). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of these amendments will not have a significant effect on the quality of the human environment (54 FR 43369).

For further details with respect to the action see (1) the application for amendments dated March 31, 1988, (2) Amendment Nos. 177, 177 , and 174 to License Nos. DPR-38, DPR-47, and DPR-55 and (3) the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC, and at the Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects I/II.

Dated at Rockville, Maryland this 16th day of November, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

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