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

ENVIRONMENT IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE

SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

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1.0 Introduction

The spent fuel storage capacity of the Fort Calhoun Station, Unit No. 1 was 178 fuel assemblies (1 and 1/3 cores) when the plant was licensed in 1973. This licensed capacity was increased in 1976 to 483 fuel assemblies (3 and 2/3 cores) by reracking the spent fuel pool. This limited increase in storage capacity was in keeping with the expectation generally held in the industry that commercial fuel reprocessing would not provide near-term relief from diminishing available storage locations.

Commercial reprocessing of spent fuel has not developed as had been originally anticipated. In 1975 the Nuclear Regulatory Commission directed 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 was to 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 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. Since the issuance of the FGEIS, applications for approximately 95 spent fuel pool capacity increases have been received and 85 have been approved. The remaining ones are still under review. 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 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 pool, the FGEIS discusses in detail other spent fuel storage alternatives. 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 various alternatives reflects 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 existing 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 only the specific environmental concerns related to the proposed expansion of the Fort Calhoun spent fuel storage capacity. This EIA consists of three major parts, plus a summary and conclusion. The three parts are: (1) descriptive material, (2) an appraisal of the environmental impact of the proposed action, and (3) an appraisal of the environmental impact of postulated accidents. 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

The currently proposed reracking plan would increase the storage capacity of the spent fuel pool from 483 fuel assemblies to 729 fuel assemblies. This would be accomplished by replacing the existing spent fuel storage racks with higher density stainless steel racks which will utilize a neutron poison material in some of the racks for reactivity control. The new rack design would also permit the implementation of disassembly and compact storage of individual rods in canisters. Although the design of the racks permits this type of storage, the licensee is proposing at this time only to expand the storage capacity to 729 whole spent fuel assemblies.

The environmental impacts associated with the operation of Fort Calhoun, as designed, were considered in the NRC's Final Environmental Statement (FES) issued in August 1972. The licensee was later authorized to increase the storage capacity from 178 to 483 assemblies. The environmental impact of that action was considered in an Environmental Impact Appraisal issued with Amendment 13 to the operating license dated July 2, 1976. In this EIA we have evaluated only additional environmental impacts which are attributable to the currently proposed increase in the spent fuel storage capacity of the plant.

1.2 Need for Increased Storage Capacity

The plant now has a licensed fuel storage capacity for 483 fuel assemblies. Currently, 265 spent fuel assemblies are stored in the pool. Assuming the current capacity, the licensee estimates that the capability to discharge the entire reactor core into the spent fuel pool would be lost in 1985. However, from a logistics point of view (removing the current racks, inserting the new racks, and also maintaining a full core discharge capability), the licensee must complete the installation before the 1984 outage, which is estimated to commence in early 1984. Thus, there is a need at this time for the licensee to make provisions to increase spent fuel storage capacity.

The use of the new racks will extend the spent fuel storage and full core discharge capability of the Station to the year 1994. The new rack design would also permit the implementation of disassembly and compact storage of individual rods in canisters beginning in 1994. This would accommodate all spent fuel through the year 2008, which is the year that the current license expires. Although the design of the racks permits this type of storage, the licensee is proposing at this time only to expand the storage capacity to 729 whole fuel assemblies. Thus, the proposed action will permit the licensee to store spent fuel and maintain a full core discharge capability to the year 1994.

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 shut down in 1972 for alterations and expansion; in September 1976, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York are licensed to store spent fuel. The storage pool at West Valley is not full, but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractual arrangements with NFS. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris operation was renewed for another 20 years; however, GE is also not accepting any additional spent fuel for storage at this facility.

2.0 Facility

The principal features of spent fuel storage at Fort Calhoun, 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 Spent Fuel Pool

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 spent fuel pool is designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. Space permiting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

The spent fuel pool is located in the auxiliary building just outside the containment. The pool is a reinforced concrete structure which is 20'-7" wide by 33'-3" long by 43'-0" deep. Wall thicknesses are 5'-6" on 3 sides and 4'-0" on one side. The floor is 12'-0" thick and is supported by steel piles which rest on sound rock. The volume of the spent fuel pool is approximately 215,000 gallons.

The pool is lined with a welded stainless steel watertight liner plate which varies in thickness from 1/16" to 3/32 inches. The liner plate is backed by and welded to a grid of stainless steel angles and plates. A leak-chase system is provided behind the liner in order to detect leaks. No modifications to the pool's physical structure will be made as a result of this proposal.

2.2 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system removes decay heat from spent fuel stored in the pool and controls and maintains the chemistry and clarity of the pool water. The system consists of two storage pool circulation pumps, a storage pool heat exchanger, a demineralizer, a filter, two fuel transfer canal drain pumps, piping, valves, and instrumentation. The pumps circulate the pool water through the heat exchanger and returns it to the pool. Cooling water to the heat exchanger is provided by the component cooling water system. The fuel transfer canal drain pumps can be used to provide make-up water to the spent fuel pool from the safety injection and refueling water tank. The purity and clarity of the pool water is maintained by diverting a portion of the circulated water through the demineralizer and the filter. There will be no change to this system as a result of the proposed modification.

2.3 Radioactive Waste Treatment System

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statement (FES) dated August 1972. There will be no change in the waste treatment systems described in Section III.D.2 of the FES because of the proposed modification.

3.0 Nonradiological Environmental Impacts of the Proposed Action

The nonradiological impacts of the Fort Calhoun Station, Unit 1, as designed, were considered in the FES issued in August 1972. No unusual terrestrial effects are anticipated or considered likely by the staff due to the proposed action. The amount of waste heat emitted by the plant as a result of the proposed action will increase slightly; however, thermal effects in the receiving water body will not be measurable by this small increase in the heat output rate. No increase in service water usage is proposed. The licensee does not propose any change in chemical usage or changes to the NPDES discharge permit. Therefore, we conclude that the Station's spent fuel pool expansion will not result in nonradiological environmental effects significantly greater or different from those already reviewed and analyzed in the FES.

4.0 Radiological Environmental Impacts of Proposed Action

4.1 Introduction

The potential radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90 are also predominantly nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to 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, however, that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP.

During and after refueling, the SFP purification system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel is cooled in the SFP and the fuel clad temperature becomes relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the licensees and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services (NFS) storage pool at West Valley, New York. Some spent fuel assemblies which had significant leakage while in operating reactors have been stored in these two pools. After storage in the onsite SFP, these fuel assemblies were later shipped to either Morris Operation or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from these fuel assemblies in the offsite storage facility.

4.2 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). Experience has demonstrated that after spent fuel decayed 4 to 6 months, there is no longer a significant release of fission products--including Kr-85--from stored fuel containing cladding defects. Forty-four (44) fuel assemblies are expected to be stored each 18 months. (Forty-five (45) fuel assemblies every third refueling.) Since space must be reserved to accommodate a complete reactor core unloading operation (nominally 133 fuel assemblies), the useful pool capacity is 729-133 = 596 fuel assemblies. At an input of 44 fuel assemblies every 18 months, it is seen that the pool has a useful storage capacity of approximately 20 years.

For the simplest case, we have assumed that all of the Kr-85 that is going to leak from defective fuel will do so in the 18 month interval between refuelings. In other words, all of the Kr-85 available for release is assumed to leak from the defective fuel before the next batch of fuel enters the pool. Our calculations show that the expected release of Kr-85 from a 44 fuel assembly refueling is approximately 52 Ci each 18 months or an average of about 35 Ci/yr. As far as potential dose to offsite populations is concerned, this is 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 with subsequent refuelings, there is no increase in the annual release of activity, 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 the pool.

In an alternative approach to determining Kr-85 releases, it can be assumed that the fission product noble gases escape over a longer period of time. If it is assumed, for example, that the fission product noble gases escape on a linear basis with time over the 20-year useful life of the pool, then only 1/20th of the Kr-85 available for release in a refueling batch would be discharged each year, or only 1.7 Ci/yr per refueling batch. On the second refueling, the release would increase by 1.7 Ci/yr for a total annual release of 3.4 Ci/yr, and so on. If natural radioactive decay is not considered, it would only be during the 20th year that the release rate of Kr-85 would reach 35 Ci/yr and then only with the SFP at full capacity. If radioactive decay is considered, the release of Kr-85 would reach only 19.6 Ci/yr during the 20th year.

As can be seen, the first approach discussed above is the most conservative. That is, the approach resulting in the greatest release of Kr-85 is based on the assumption that all of the Kr-85 available for release from failed fuel in a single reload is discharged prior to the next reload operation. In such a calculation, the enlarged capacity of the SFP has no effect on the total amount of Kr-85 released each year.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 (I-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 I-131 inventory in the fuel will decay to negligible levels between refuelings.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above the value of 120°F during normal refuelings used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed 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 SFP. Therefore, even if there were a higher evaporation rate from the SFP, the increase in tritium and iodine released from the plant, as a result of the increase in 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 release of gaseous activity.

We have evaluated the additional dose to the public as a result of the SFP modification. The additional total body dose that might be received by an individual at the site boundary and the estimated dose to the total body of the population within a 50-mile radius of the plant is less than 0.10 mrem/yr and 0.01 person-rem/yr, respectively. These doses are small compared to the fluctuations in the annual dose this population receives from exposure to background radiation. The population dose represents an increase of less than 0.01 percent of the dose from the plant evaluated in the FES.

4.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filter and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filter 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 to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we agree with licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two demineralizer resin beds a year due to the increased operation of the spent fuel pool cleanup system. The annual average volume of solid wastes shipped from the Fort Calhoun site during 1976 through 1981 was 16,700 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 100 cubic feet per year, the increase in total waste volume shipped would be less than 1% and would not have any significant additional environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification are contaminated and will be disposed of as low level solid waste. We estimate that approximately 12,000 cubic feet of solid radwaste will be removed from the plant because of the proposed modification. This estimate is based on the licensee's estimate that the weight of the solid waste removed will be 158,000 pounds. Averaged over the lifetime of the plant, this would increase the total waste volume shipped from the facility by less than 4%. This will not have any significant additional environmental impact.

4.4 Radioactive Material Released to Receiving Waters

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

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

4.5 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation dose. The occupational dose for the operation is estimated by the licensee to be 3 person-rems based on the licensee's detailed breakdown of occupational dose for each phase of the modification. In making this estimate, the licensee considered the number of individuals performing this job, their occupancy time while performing the job, and the average dose rate in the area where the job is being performed. This dose is a small fraction of the total annual person-rems from occupational dose for all plant operations.

We have estimated the increase in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the SFP area from radionuclide concentrations in the SFP water and from the SFP assemblies. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add only a small fraction to the total annual occupational radiation dose at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in dose received by workers.

5.0 Environmental Impacts of Postulated Accidents

5.1 Cask Drop Accident

The licensee states that it does not presently utilize or own a spent fuel sample cask or spent fuel shipping cask. If such casks were to be utilized, the single failure-proof main hook of the Auxiliary Building Crane would be used to transport these casks. This characteristic, in conjunction with the presence of electrical interlocks which preclude travel of the main hook and auxiliary hook over the spent fuel pool (NUREG-0612), would greatly reduce the likelihood of occurrence of a cask drop, obviating the need for consideration of radiological release impact of such an accident. Therefore, we conclude that an analysis of the radiological consequences of a cask drop accident is not required.

5.2 Spent Fuel Pool Gate Drop Accident

The licensee states that the non-single-failure-proof 10-ton auxiliary hook of the Auxiliary Building Crane is used to move the spent fuel pool (SFP) gate. The gate is moved directly from its notch in the spent fuel pool wall to storage in the transfer canal without moving over spent fuel. If the gate were dropped and were to undergo a highly unlikely rotational motion following swinging of the supporting wire rope and hook, it could impact the tops of the spent fuel racks. The impact kinetic energy transmitted to a particular cell is no more than that transmitted in a fuel handling accident, therefore offsite radiological consequences would not exceed those from such an accident.

5.3 Fuel Handling Accident

The licensee has proposed to expand the storage capacity of the SFP from 483 spent fuel assemblies to 729 assemblies. During the action, the maximum weight of loads which may be transported over spent fuel in the pool will be limited to that of a single assembly by an excess weight interlock on the fuel lifting hoist (LCO 2.8, Amendment No. 24 to Fort Calhoun Station FSAR, June 6, 1977). The proposed spent fuel pool modification does not, therefore, increase radiological consequences of fuel handling accidents considered in the staff FES of August, 1972, since this accident would still result in, at most, release of the gap activity of one fuel assembly due to the limitations on available impact kinetic energy.

5.4 Conclusions

Based upon the above evaluation, the staff concludes that the likelihood of a cask drop accident resulting in radionuclide releases is sufficiently small that this accident need not be considered. Also, if a highly unlikely SFP gate drop accident should occur, we believe that radionuclide releases no greater than those postulated in a fuel handling accident would result. Additionally, a fuel handling accident would not be expected to result in radionuclide releases leading to offsite radiological consequences exceeding those of the fuel handling accident in the staff FES of August 1972. Our present analysis indicates a 0-2 hr. Exclusion Area Boundary (EAB) Whole Body dose of 11.5 millirem. These estimated doses are well within the 10 CFR Part 20 guideline values. Therefore, we conclude that the proposed modifications are acceptable.

6.0 Summary

The Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel concluded that the environmental impact of interim storage of spent fuel was negligible and the cost of the various alternatives reflects the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in SFP designs the FGEIS recommended licensing SFP expansion on a case-by-case basis. For Fort Calhoun, the expansion of the storage capacity of the SFP will not create any significant additional radiological effects. The additional total body dose that might be received by an individual at the site boundary and the estimated dose to the total body of the population within a 50-mile radius of the plant is less than 0.10 mrem/yr and 0.01 person-rem/yr, respectively. These doses are small compared to the fluctuations in the annual dose this population receives from exposure to background radiation. The population dose represents an increase of less than 0.01 percent of the dose from the plant evaluated in the FES. The occupational radiation dose or workers during the modification of the present storage racks is estimated by the licensee to be 3 person-rem. This is a small fraction of the total person-rems from occupational dose at the plant. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational dose within the limits of 10 CFR Part 20, and as low as reasonably achievable.

7.0 Basis and Conclusion for Not Preparing an Environmental Impact Statement

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51 and the Council on Environmental Quality's Guidelines, 40 CFR 1500.6. We have determined, based on this assessment, that the proposed license amendment 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.

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8.0 References

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3. OPPD letter to USNRC dated June 23, 1982.
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