SAFETY EVALUATION

BY THE

DIVISION OF OPERATING REACTORS

CONCERNING

DUKE POWER COMPANY'S

APPLICATION FOR AMENDMENT TO ITS OPERATING LICENSE

TO INCREASE THE AUTHORIZED CAPACITY OF THE SPENT FUEL POOL

AT THE

OCONEE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-269 AND 50-270

1.0 INTRODUCTION

By its letter dated February 2, 1979 the Duke Power Company (DPC) applied for a license amendment to increase the authorized storage capacity for spent fuel at the Oconee Nuclear Station, Units 1 and 2, from 336 to 750 fuel assemblies.

2.0 DISCUSSION

The proposed spent fuel racks are to be made up of individual containers which are approximately 9 inches square by 16 feet long. These containers are to be fabricated from 0.250 inch thick, type 304 stainless steel. The rack structure is designed to hold these square containers on a 13.75 inch pitch under safe shutdown earthquake accelerations. Thus, there will be over three inches of water between neighboring containers. The 13.75 inch pitch combined with the overall dimension of the fuel assembly, which is 8.52 inches, gives a fuel region volume fraction of 0.38 for the storage lattice.

DPC states that the highest anticipated U-235 enrichment is 3.5 percent. This value was used in the neutron multiplication factor calculations. This enrichment in the present fuel assemblies results in a fuel loading of 46.0 grams of U-235 per axial centimeter of fuel assembly.

2.1 CRITICALITY ANALYSES

As stated in DPC's February 2, 1979 submittal, the fuel pool criticality calculations are based on unirradiated fuel assemblies with no burnable poisons which have a fuel enrichment of 3.5 weight percent U-235. This

corresponds to a fuel loading of 46.0 grams of U-235 per axial centimeter of these fuel assemblies. For the criticality calculations it was also assumed that the water in the pool was pure, i.e., unborated.

Combustion Engineering's CEPAK computer program was used to get the multi-group cross sections for the criticality analysis. The NUTEST computer program was used to calculate the self shielding and flux advantage factors for the material heterogeneity, and the DOT-2W discrete ordinates transport program was used for the overall storage lattice cell calculations. These computer programs were first used to calculate the neutron multiplication factor for an infinite array of fuel assemblies in the nominal storage lattice. The maximal effects of the stainless steel thickness tolerance, fabrication tolerances, fuel assembly positioning uncertainties, and water temperature were then calculated. The accuracy of these methods was checked by calculating the following sets of experiments:

1. The criticality of five, cold, clean PWR's.

2. Stainless steel clad UO_2-H_2O lattice experiments,

3. The LaCrosse Boiling Water Reactor critical experiments with stainless steel shrouds.

4. The reactivity worths of stainless steel reflectors on a uranyl fluoride solution reactor.

The results of these calculations indicate that the total uncertainty in the storage lattice cell calculations might be as large as 1.8 percent Δk; so DPC allowed this amount of margin in the design.

2.1.1 EVALUATION

The above described results compare conservatively with the results of parametric calculations made with other methods for similar fuel pool storage lattices. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

We find that all factors that could affect the neutron multiplication factor in this pool have been conservatively accounted for and that the maximum neutron multiplication factor in this pool with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a spent fuel pool. This 0.95 acceptance criterion is based on the uncertainties associated

with the calculational methods and provides sufficient margins to preclude criticality in the fuel. Accordingly, there is a technical specification which limits the effective neutron multiplication factor in the spent fuel pool to 0.95.

2.1.2 CONCLUSION

We find that when any number of the fuel assemblies, which DPC described in these submittals, having no more than 46.0 grams of uranium-235 per axial centimeter of fuel assembly or equivalent are loaded into the proposed racks, the keff in the fuel pool will be less than the 0.95 limit. We also find that in order to preclude the possibility of the keff in the fuel pool from exceeding this 0.95 limit without being detected, the use of these high density storage racks will be prohibited for fuel assemblies that contain more than 46.0 grams of uranium-235, or equivalent, per axial centimeter of fuel assembly. On the basis of the information submitted, and the keff and fuel loading limits stated above we conclude that the health and safety of the public will not be endangered by the use of the proposed racks.

2.2 SPENT FUEL COOLING

The licensed thermal power for each of Oconee units one and two is 2568 MWth. DPC plans to refuel these reactors every eighteen months at which times about 70 of the 177 fuel assemblies in the cores will be replaced. To calculate the maximum heat loads in the spent fuel pool DPC assumed a 168 hour time interval between reactor shutdown and the time when either the 70 fuel assemblies in the normal refueling or the 177 fuel assemblies in the full core offload are placed in the spent fuel pool. For this cooling time DPC used the method given in the NRC Standard Review Plan 9.2.5 to calculate maximum heat loads of 19.6×10^6 BTU/hr for a normal refueling and 31.7×10^6 BTU/hr for a full core offload.

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

DPC states that this sytem will be sufficient to keep the spent fuel pool water temperature below 150°F until the first quarter of 1980 when an additional spent fuel pool cooling pump and heat exchanger of the same capacity will be installed.

2.2.1 EVALUATION

Using the method given on pages 9.2.5-8 through 14 of the November 24, 1975 version of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^7 seconds, we calculate that the maximum peak heat load during the refueling which would fill the pool could be 20 x 10^6 BTU/hr and that the maximum peak heat loads for a full core offload that essentially fills the pool could be 34 x 10^6 BTU/hr. This full core offload was assumed to be a fully irradiated core which was taken out of its reactor vessel 35 days after the other Oconee unit, which shares this spent fuel pool, had been refueled. We also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pool from 336 to 750 is 1.9 x 10^6 BTU/hr. This is the difference in peak heat loads for the present and the modified pools.

We find that with the three pumps operating, as DPC has committed to provide by the first quarter of 1980, the cooling system can maintain the fuel pool outlet water temperature below 125°F for the normal refueling offload that fills the pool and below 136°F for the full core offload that fills the pool. In the highly unlikely event that all three spent fuel pool cooling systems were to fail at the time when there was a peak heat load from a full core in the pool, we calculate that the maximum heat up rate of the spent fuel pool water would be 9.0°F/hr. Thus, if the water were initially at an average temperature of 125°F it would be more than nine hours before boiling would start. We also calculate that after boiling starts the required water make up rate will be less than 70 gpm. We find that nine hours will be sufficient time to establish a 70 gpm make up rate.

2.2.2 CONCLUSION

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

2.3 INSTALLATION OF RACKS AND FUEL HANDLING

In their February 2, 1979, proposal DPC states that at the time of the installation of the new racks there will be 140 spent fuel assemblies in the pool. Initially, these will all be placed in existing racks at the

south end of the pool. This will allow the removal of approximately one third of the existing racks, which are at the north end of the pool, and the installation of two new racks without getting close to the spent fuel. For the installation of the rest of the racks DPC has developed a detailed procedure for redistributing the 140 fuel assemblies between the south end of the pool and the new racks in the north end of the pool so there will be a minimum of fourteen feet of open space between the work area and racks with fuel in them. Also, the plan is to move the racks in the pool at an elevation which is lower than the top of any stored fuel assemblies.

2.3.1 EVALUATION

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

2.3.2 CONCLUSION

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.