SAFETY EVALUATION BY THE
DIVISION OF OPERATING REACTORS
CONCERNING
COMMONWEALTH EDISON'S
APPLICATION FOR AMENDMENT TO ITS OPERATING LICENSE
TO INCREASE THE AUTHORIZED CAPACITY OF THE SPENT FUEL POOL
AT THE
DRESDEN NUCLEAR POWER PLANT UNITS 2 AND 3
DOCKET NUMBERS 50-237 AND 50-249

1.0 INTRODUCTION

By its letter dated May 11, 1978 as supplemented by letter dated January 12, 1979, Commonwealth Edison (CE) applied for a license amendment to increase the authorized storage capacity for spent fuel at the Dresden Nuclear Power Plant Units 2 and 3 from 2840 to 7560 fuel assemblies by installing new racks in both of the spent fuel pools.

2.0 DISCUSSION

The proposed spent fuel assembly racks are to be made up of alternating stainless steel containers. Thus, there will be only one container wall between adjacent spent fuel assemblies. Each container wall is to have a sheet of Boral sandwiched between two stainless steel plates. The containers will be about 14 feet long and will have a square cross section with an outer dimension of 6.384 inches and a total wall thickness of 0.217 inches. The nominal pitch between fuel assemblies will be 6.3 inches. The overall fuel region volume fraction for 8 x 8 fuel assemblies in the proposed stroage racks is 0.652. The Boral sheet consists of a central core of a 0.056 inch thick dispersion of boron carbide in aluminum. This central core is clad on both sides with 0.010 inches of aluminum. CE states that the minimum homogeneous concentration of the boron-ten isotope will be 0.02 grams per square centimeter of the Boral plate. This is equivalent to 1.2×10^{21} boron-ten atoms per square centimeter.

2.1 Criticality Analyses

CE's fuel pool criticality calculations are based on unirradiated fuel assemblies with no burnable poison and a maximum fuel loading of 14.8 grams of uranium-235 per axial centimeter of fuel assembly. These calculations were made by the Nuclear Services Corporation (NSC) for CE. The basic method was to use the CHEETAH and XSDRN computer programs to generate four energy group cross sections for use in the CITATION diffusion program. XSDRN, which is a one-dimensional, discrete-ordinates, spectral-averaging program was used to calculate the cross sections for the Boral core regions. Also, the internal black boundary condition in the CITATION program was used to calculate the neutron flux in the thermal energy group in the Boral plates. NSC checked the accuracy of this calculational method by using it to calculate two critical experiments which had Boral plates in them. As shown in Table 3.3-3 of the licensee's submittal, the resulting neutron multiplication factors were more than one percent higher than the experimentally determined values. Thus, NSC assumed that this calculational method gives conservative results for the neutron multiplication factor in the spent fuel pool.

NSC first used these programs to calculate a neutron multiplication factor, k_∞, of 0.91 for the nominal proposed storage rack lattice while assuming that the density of the boron ten in the Boral was at its minimum value of 0.02 grams per square centimeter of Boral plate and the pool temperature was 40°F.

NSC then calculated the neutron multiplication factor for each of the following conditions: (1) increasing the temperature to 212°F, (2) increasing the lattice pitch (3) eccentrically positioning the fuel assemblies in the racks, and (4) taking the Zircaloy channels off of the fuel assemblies which are placed in these racks. NSC found that all of these changes decreased the neutron multiplication factor in the pool. NSC then calculated the following possible increases in the neutron multiplication factor (Δk):

1. One extra fuel assembly at the outer periphery of the rack ---+.002
2. All of the racks pushed as close together as possible --------+.018
3. One out of every 32 Boral plates missing --------------------+.015

2.1.1 Evaluation

A comparison of the above results with the results of other calculations which were made for high density, spent fuel, storage lattices with boron plates, shows them to be acceptably accurate.

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.

The NRC acceptance criteria for the criticality aspects of high density fuel storage racks is that the neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions throughout the life of the racks. This 0.95 acceptance criterion is based on the overall uncertainties associated with the calculational methods, and it is our judgement that this provides sufficient margin to preclude criticality in fuel pools. Accordingly, there is a technical specification which limits the neutron multiplication factor, k_{eff}, in spent fuel pools to 0.95. Since the neutron multiplication factor in spent fuel pools is not a quantity which is measured with good accuracy, the only available value is a calculated one. To preclude any unreviewed increase, or increased uncertainty, in the calculated value of the neutron multiplication factor which could raise the actual k_{eff} in the fuel pool above 0.95 without being detected, a limit on the maximum fuel loading is also required. Accordingly, we find that the proposed high density storage racks will meet the NRC criteria when the fuel loading in the assemblies described in these submittals is limited to 14.8 grams or less of uranium-235 per axial centimeter of fuel assembly.

In its response to our request for additional information CE stated that in addition to the usual quality assurance program, a neutron poison verification test will be conducted at the Dresden plant after the racks are installed in the pool. This will be a test to statiscally show with 95 percent confidence that the boron is not missing from more than one out of every thirty two Boral plates. We find that this will not cause the neutron multiplication factor in the fuel pool to increase above 0.95. However, in this test, if any Boral plates are found to be missing the NRC shall be notified and a complete test on every storage location shall be performed.

2.1.2 Conclusion

We find that when any number of the fuel assemblies, which CE described in these submittals, which have no more than 14.8 grams of uranium-235 per axial centimeter of fuel assembly, are loaded into the proposed racks, the k_{eff} in the fuel pool will be less than the 0.95 limit. We also find that in order to preclude the possibility of the k_{eff} in the fuel pool from exceeding this 0.95 limit without being detected, it is necessary, pending NRC review, to prohibit the use of these high density storage racks for fuel assemblies that contain more than 14.8 grams of uranium-235 per axial centimeter of fuel assembly. On the basis of the information submitted, and the k_{eff} 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 unit of Dresden II and III is 2527 MWT. CE has been refueling these units annually, but in the future it plans to extend the refueling periods to eighteen months. For each eighteen month refueling CE assumed that about forty two percent, or 306, of the 724 assemblies in the core would be moved to the spent fuel pool in the ten day time period following the shutdown of the reactor. For the purpose of determining the maximum possible heat load CE also assumed that a full core, i.e., 724 assemblies, could be moved to the spent fuel pool in the ten day time period following the shutdown of the reactor. For the power history prior to refueling, CE assumed an energy production of 19,000 MWD/MTU obtained at a continuous energy density of 20 MW/MTU. With these assumptions CE used the method given in American National Standard 5.1 to calculate 11.3×10^6 BTU/hr as the maximum heat load for any refueling and 22.6×10^6 BTU/hr as the maximum heat load for a full core offload.

The spent fuel pool cooling system as described in Chapter 10 of the FSAR, consists of two pumps and two heat exchangers. Each pump is designed to pump 700 gpm (3.5×10^5 pounds per hour) and each heat exchanger is designed to transfer 3.65×10^6 BTU/hr from 125^oF fuel pool water to 105^oF Reactor Building Cooling Water, which is flowing through the heat exchanger at a rate of 7.5×10^5 pounds per hour.

When a full core is discharged to the spent fuel pool one of the three loops of the Shutdown Reactor Cooling System will be connected in parallel with the Fuel Pool Cooling System. This connection will provide an additional flow of 3000 gpm (1.5 x 10^6 BTU/hr) of fuel pool water which will be cooled by the Shutdown Reactor Cooling System.

2.2.1 Evaluation

Using the method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^3 seconds, we calculate that the maximum peak heat load during the thirteenth refueling could be 13.2 x 10^6 BTU/hr and that the maximum peak heat load for a full core offload that fills the pool could be 26.2 x 10^6 BTU/hr. This full core offload was assumed to take place one and one half years after the eleventh refueling. 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 1420 to 3780 will be 2.1 x 10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools.

We calculate that with both pumps operating, the spent fuel pool cooling system can maintain the fuel pool outlet water temperature below 141°F for a peak refueling heat load of 13.2 x 10^6 BTU/hr. We find that when the Shutdown Reactor Cooling System is connected in parallel with the spent fuel pool cooling system, the combined system will have sufficient capacity to keep the spent fuel pool outlet water temperature below 145°F for a full core heat load of 26.2 x 10^6 BTU/hr.

Assuming a maximum fuel pool temperature of 145°F the minimum possible time to achieve bulk pool boiling after any credible accident will be about eight hours. After bulk boiling commences, the maximum evaporation rate will be 54 gpm. We find that eight hours would be sufficient time for CE to establish a 54 gpm make up rate. We also find that under bulk boiling conditions the temperature of the fuel will not exceed 350°F. This is an acceptable temperature from the standpoint of fuel element integrity and surface corrosion.

2.2.2 Conclusion

We find that the present cooling capacities of the Dresden 2 and Dresden 3 systems are sufficient to handle the incremental heat loads that will be added by the proposed modifications. We also find that these incremental heat loads will not alter the safety considerations of spent fuel pool 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

The fuel building crane, which will be used to remove the present racks and install the new ones, is rated for a 125 ton load. The heaviest rack will weigh about six tons. CE states in their submittal that the racks will not be carried over stored fuel assemblies. To prevent this, CE plans to move the fuel assemblies which are now in the pool, as far away as possible from the location where the racks are being changed; i.e., to the other end of the pool.

2.3.1 Evaluation

At the beginning of 1979 there were 1069 fuel assemblies stored in the Dresden 2 and Dresden 3 pools. Since the present capacity is 2840 assemblies this means that fuel assemblies can be removed from over one half of the racks in the pools. After the 1979 refuelings there will be less room, but we find that there will still be enough room to allow the replacement of the racks without having to move them over fuel assemblies.

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.