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January 24, 1979

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 & 3
Response to Request for Additional Information for Spent Fuel Pool Modification
NRC Docket Nos. 50-237 and 50-249

Reference (a): Robert F. Janecek letter to Director of Nuclear Reactor Regulation dated January 12, 1979

Dear Sir:

Reference (a) transmitted our responses to a request for additional information for the subject modification. Our response to Question 1 erroneously reported the total expected dose as "1.86 to 46.8 Man-Rem." The correct values should read "18.7 to 46.8 Man-Rem."

Additionally, it has come to our attention that several of the responses did not copy well enough to read clearly. Enclosed are additional copies of responses to Questions 5, 8, 10, 11, 12, 13, and 14 which should replace the previously transmitted pages.

Please direct any additional questions or comments to this office.

One (1) signed original and thirty-nine (39) copies of this letter are provided for your use.

Very truly yours,

Robert F. Janecek

Robert F. Janecek
Nuclear Licensing Administrator
Boiling Water Reactors

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QUESTION NUMBER 5:

Provide the estimated volume of contaminated material (e.g., spent fuel racks, seismic restraints) expected to be removed from the spent fuel pools during the modification and shipped from the plant to a licensed burial site.

RESPONSE:

The as-built volume of the spent fuel racks is about 19,500 ft^3. The minimum volume or the volume of the metal in the racks is approximately 1,600 ft^3.

QUESTION NUMBER 8:

Your May 11, 1978 submittal did not address the impact of the proposed SFP modification on the environment. Discuss in some detail the impact of the proposed SFP modification on the following:

a. radioactive gaseous effluents from the pool,
b. radioactive liquid effluents from the plant, including leakage of water from the pool and the SFP leak collection system, and
c. radioactive solid wastes from the plant, including the change in the frequency of replacing the filter-demineralizer resin and the volume of the resin bed.

RESPONSE:

a. The amount of radioactivity in the spent fuel storage pool following modification will differ only slightly from that in the current pool. One year after shutdown the amount of radioactivity in spent fuel is less than 1% of its value at shutdown due to radioactive decay. After cooling for two more years, the radioactivity is reduced to about 0.3% of its initial value. Ten years after shutdown, it decreases further to slightly over 0.1%. Thus, the great majority (say >85%) of the radioactivity will come from fuel just discharged. Therefore, only a very slight change in the amount of radioactivity, and the corresponding dose rates at the pool surface will be a result of expanding the spent fuel pool storage capacity. Since gaseous effluents like Kr-85 and Xe-133 contribute only a fraction of the total radioactivity, the impact of the modification on radioactive gaseous effluents will be negligible.

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QUESTION NUMBER 10:

The proposed technical specification limit on k_{eff} (5.5 B.) is, by itself, not sufficient for high density spent fuel storage racks. Since the k_{eff} in spent fuel pools is a quantity which is not 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 fuel loading is also required. This limit can be imposed in either of two ways. The first way is by specifying the numbers of grams of uranium-235 per axial centimeter of the fuel assemblies that were used for the calculations in the Licensing Report and then making both this Licensing Report and the NRC's Safety Evaluation of this report the basis for the k_{eff} in Technical Specification 5.5 B. The second way is by directly limiting future fuel loadings in assemblies that are placed in these high density racks to the maximum number of grams of uranium-235 per axial centimeter of assembly that was used in these calculations. Provide your proposed limit on fuel loading.

RESPONSE:

Two types of fuel assembly designs have been considered in the keff calculations. Relevant parameters are shown in Table 1. The corresponding numbers of grams of U^{235} per axial centimeters of the fuel assemblies are calculated as follows:

Continued.......

RESPONSE (Continued):

For the first type of fuel assembly

	Number of fuel rods	= 49
	Fuel enrichment	= 2.8 wt % U^{235}
∴	Density of U^{235}	= $(0.93 \times 10.96 \times 0.028)(\frac{235}{235+2\times16})$
		= 0.251 gm/cc
	Number of grams of U^{235} per axial centimeter of fuel assemblies	= 0.251 x total area of fuel rods
		= $0.251 \times (\frac{0.488\times2.54}{2})^2 \times \pi \times 49$
		= 14.84 gm/cm

For the second type of fuel assembly

	Number of fuel rods	= 62
	Fuel enrichment	= 3.01 wt % U^{235}
∴	Density of U^{235}	= $(0.95 \times 10.96 \times 0.0301)(\frac{235}{235+2\times16})$
		= 0.276 gm/cc
	Number of grams of U^{235} per axial centimeter of fuel assemblies	= 0.276 x total area of fuel rods
		= $0.276 \times (\frac{0.41\times2.54}{2})^2 \times \pi \times 62$
		= 14.58 gm/cm

The following remarks are made concerning the limit on fuel loading:

1. The fuel enrichment used in the calculations are the highest average assembly enrichments to be cycled in the plant.

2. The computer codes used in the calculations are production codes widely used and thoroughly tested by the nuclear industry. The licensing report includes comparison of CHEETAH-XSDRN-CITATION code results with two benchmark critical experiments involving boral plates. The positive bias demonstrated by these calculations is not even taken into consideration in arriving at keff $\leq$ 0.95 for extra margin of conservatism.

3. The keff values of 0.91788 and 0.92096 calculated for the two types of fuel assembly designs are still very far below 0.95 from the criticality point of view.

TABLE 1

FUEL DESIGN PARAMETERS

Fuel Assembly Array	7 x 7	8 x 8
No. of Fuel Rods	49	62
No. of Water Rods	0	2
Fuel Pitch	0.738"	0.64"
Fuel Pellet O.D.	0.488"	0.41"
Clad O.D.	0.563"	0.483"
Clad Thickness	0.032"	0.032"
Clad Material	Zircaloy 2	Zircaloy 2
Pellet Density, % T.D.	93.0	95.0
Max. Bundle Enrichment, wt % U^{235}	2.80	3.01
Nominal Active Fuel Length	144"	144"
Channel Material	Zircaloy 4	Zircaloy 4
Channel O.D.	5.438"	5.438"
Channel Thickness	0.080"	0.080"
Water Rod O.D.	-	0.591"
Water Rod Thickness/Material	-	0.030"/Zircaloy 2

Theoretical density of UO_2 = 10.96 gm/cc

QUESTION NO. 11:

State the density of boron ten atoms in region 7 of Figure 3.3-1 and state the minimum areal density of boron ten atoms in the Boral plates as will be certified by quality control records.

RESPONSE:

A minimum areal density of 0.02 grams of boron ten per square centimeter of Boral plate was used in the calculations.

The manufacture of Boral and fabrication into plates is controlled by the Brooks and Perkins Quality Assurance Program, which includes detailed procedures for the inspection and verification of boron ten loading in each Boral plate. The inspection plan for these activities will include the following:

1. Documented laboratory analysis for chemical boron content and isotopic boron ten content of each batch of boron carbide powder.

2. Inspector's verification of the weighing and mixing of boron carbide and aluminum powder into a batch, according to the production plan, and the assembly of this batch into one or several ingots, all identified and traceable to the batch.

3. Documented laboratory analysis of a selected sample of batch mixes to verify boron carbide content.

Continued........

RESPONSE (Continued):

4. The rolling of the ingot into a sheet, and the subsequent blanking of a sheet into two or three plates, each identified and traceable to the ingot and batch.

5. Visual inspection of the perimeter of each plate to verify that the core extends to the edge (the aluminum edge filler has been completely sheared away), and a check of plate thickness at several points.

6. Documented laboratory analysis, according to a sampling plan, of the boron carbide content of coupons cut from each end of each plate.

7. Documented neutron transmission tests over the surface of a selected sample of plates to verify the uniformity of boron ten loading across the entire plate area.

The detailed sampling plans will be established prior to manufacturing, based on the specific production lot sizes to be used in order to establish a 95 percent confidence that the minimum areal density of 0.02 grams of boron ten per square centimeter is present over the entire area of each plate.

It is estimated that the nominal level of boron ten areal density will be 0.0220 $\pm$ 0.0012 grams per square centimeter, to provide an assured minimum level of 0.02. The exact range will be established in the detailed production plans.

QUESTION NUMBER 12:

Provide a description of the onsite test you intend to perform to verify, within 95 percent confidence limits, that a sufficient number of Boral plates in the installed racks will contain the required boron content to maintain the $k_{eff} \leq 0.95$.

RESPONSE:

A neutron poison verification test will be conducted at the Dresden Plant after the racks are installed in the pool. This will be a qualitative test to verify the fuel storage locations contain neutron absorber material in each wall of the cell as fabricated. Sufficient storage cells of each rack will be verified following installation to verify, within 95 percent confidence, that Boral plates are in the installed racks.

This procedure is similar to the poison verification tests conducted at Montecello and TVA by National Nuclear Corporation utilizing their proprietary equipment.

QUESTION NUMBER 13:

Provide the dimensions and tolerances on the rounded corner containers that show that the pitch of this alternating lattice cannot be less than 6.3 inches anywhere in the pool.

RESPONSE:

During assembly of the racks, the final assembly fixture maintains the correct absorber tube center-to-center spacing within plus/minus 1/16 inch. Therefore, the fuel storage lattice pitch tolerance is less than plus/minus 1/16 inch of the nominal 6.3 inch. An allowance for a lattice pitch tolerance of plus/minus 0.100 inch has been included in the nuclear criticality analysis.

QUESTION NUMBER 14:

Specify the design flow rate of the reactor building cooling water through the spent fuel pool heat exchangers and the maximum temperature of this water going into these heat exchangers.

RESPONSE:

The design flow rate of cooling water to the heat exchangers is 1500 gpm per heat exchanger. The maximum inlet temperature is 105°F.