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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Provides addl info re spent fuel storage capacity in response to 790102 request.Response covers:distribution of sizes & median size of boron carbide particles in Boron plate & venting of stainless steel envelopes.

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January 24, 1979
AEP:NRC:00128

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
Additional Information on the Spent Fuel Storage Capacity Expansion

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

The attachment to this letter provides additional information on the spent fuel storage capacity expansion program for the Donald C. Cook Nuclear Plant. This information was requested by Mr. A. Schwencer, in his letter dated January 2, 1979 as a result of the staff review of our November 22, 1978 submittal.

Very truly yours,

John Tillinghast
Vice President

JT:em

Sworn and subscribed to before me
this 24 day of January, 1979 in
New York County, New York

Notary Public

cc: R. C. Callen
G. Charnoff
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R. J. Vollen
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PLANT SYSTEMS BRANCH

REQUEST NO. 1

Provide the distribution of sizes and the median size of the boron carbide particles in the Boral plate which was used in the "Bierman" (PNL-2438) critical particles which are specified as Items 5 and 6 on Table 3.1.8 of your November 22, 1978 submittal "Spent Fuel Storage Capacity Expansion".

RESPONSE:

The B_4C particles are sixty (60) mesh and finer with a mean particle size of approximately one-hundred and seventy-five (175) mesh.

REQUEST NO. 2

Provide the distribution of sizes and the median size of the boron carbide particles in the Boral plates in the proposed racks. Also, describe how the self-shielding of the boron carbide particles is accounted for in determining the effective cross section of the Boral plates.

RESPONSE:

The Boral plates proposed for use in the Donald C. Cook spent fuel storage racks contain the same B_4C particle size and distribution as the Boral plates described in the previous answer (i.e., the "Bierman" critical experiments).

Accordingly, the KENO benchmark analyses performed by the Exxon Nuclear Company are based on fuel, lattice geometries and poison materials closely simulating conditions for which the Donald C. Cook Nuclear Plant criticality analyses were performed.

The Exxon Nuclear Company analyses were performed by comparing the effect of lumping the B_4C contained in each Boral plate into thin slabs of pure B_4C versus the effect of homogeneously distributing the B_4C and aluminum into a uniform thicker slab. The resultant change in K_{eff} due to the difference in self-shielding was $\leq$.005 (i.e., within the statistical uncertainty of the KENO analyses). In addition to the above analyses, an 18 group versus a 123 group KENO analysis was performed comparing the spatial lumping effects described above with similar results.

In summary, the KENO benchmark analyses referenced in the original license submittal and the comparison of analytical models and methodology described above insure the appropriateness of the Donald C. Cook Nuclear Plant criticality analyses.

REQUEST NO. 3

If you do not propose to vent the stainless steel envelopes containing Boral plates, state the maximum K_{eff} in the fuel pool that could be obtained from the following scenario. Leaks occur near the bottom of the Boral envelopes of several adjacent guide tubes which allow pool water to enter the envelopes and chemicallyreact with the aluminum coating on the Boral to form aluminum oxide and hydrogen. The resulting generation of hydrogen causes the stainless steel envelopes of several adjacent guide tubes with fuel assemblies in them to swell.

RESPONSE:

We will implement a set of fabrication, process, and QA/QC controls which insure that the completed storage cells for the D. C. Cook Nuclear Plant Project are leak tight to at least a 95/95 statistical level. Accordingly, the likelihood that any storage cell will leak, let alone develop a leak which results in swelling, is extremely small.

The Exxon Nuclear Company has performed however, detailed KENO analyses for the unlikely case of one (1) storage cell in a 5x5 array which does leak and swell in pure, unborated water in such a fashion as to maximize K_{eff}. This unlikely case has been shown to increase K_{eff} by $\leq$1.5% and still meet the $\leq$0.95 criteria.

The Exxon Nuclear Company has also determined that the consequences of a non-credible or faulted number of adjacent leaking fuel cells would result in a $K_{eff} \leq 0.85$ in a borated ($\leq$1700 ppm) pool water environment.*

REQUEST NO. 4

Section 3.1.4 which is entitled "Storage Array Description" does not state whether or not there will be enough space between the periphery of the racks and the pool walls to insert a fuel assembly. If there is, provide the maximum K_{eff} that will be obtained when a fuel assembly is brought up as close as possible to a rack which is fully loaded with fuel assemblies.

RESPONSE:

The maximum K_{eff}, that can be obtained when a fuel assembly is brought as close as possible to the portion of a rack which is fully loaded with fuel assemblies, has been determined to be less than the maximum internal rack module array reactivity. This is due to the additional 2-inches of water and 1/8-inch of steel that separates any fuel assembly brought between the pool wall and a fully loaded rack. The additional water and steel separation are the result of structural members included on the periphery of rack modules.

* U.S.N.R.C. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", Paragraph 1.2, April 14, 1978.

REQUEST NO. 5

The NRC requires an on-site neutron attenuation test to verify the presence of the boron. This is in addition to the Quality Assurance Program you described in Section 3.4.1. Provide a description of the neutron attenuation test that you will perform at the Cook Site to statistically show with 95 percent confidence that the boron is not missing from enough plates to allow the K_{eff} to go above 0.95.

RESPONSE

Stringent in-process inspection and process controls are imposed during manufacturing of the Boral to assure that the Boral plates contain at least 0.020 gms B-10/cm^2. Additional measures assure that the plates are properly inserted in each completed storage cell. Accordingly, the on-site neutron attenuation test is only used to supplement the other controls and tests used to insure the presence of boron.

In addition to these controls, KENO-IV analyses were performed with one Boral poison plate missing in a 5 x 5 storage cell array resulting in a K_{eff} of 0.928 ± 0.004. Consequently, the design assures that the K_{eff} criteria of 0.95 or less is met for all conditions previously delineated in the license amendment submittal as well as for the unlikely event of one Boral plate missing.

The tests will be performed by inserting a shielded moderated neutron source into one storage cell and a neutron detector into an adjacent storage cell. Prior to the actual tests, calibration tests will be performed which will provide the necessary counting statistics for demonstrating whether any Boral plates are missing or do not contain the boron poison material.

The tests will be performed on the following basis:

1. A 100% storage cell inspection of the first rack module.
2. A 10% random sampling of the storage cells in the remaining rack modules.
3. Should any one measurement demonstrate that a Boral plate is missing or does not contain the boron poison material, a 100% inspection program will be conducted on all rack modules.

A control procedure will be prepared and utilized during the measurements and a permanent record will be generated and retained for each measurement.

REQUEST NO. 6

There appears to be a deficiency in the boron surveillance program described in Section 3.4.2. In order for this test with sealed samples to be valid, the probability of developing a leak in the surveillance sample must be the same as the probability of developing a leak in the Boral envelopes in any of the racks. These two probabilities do not appear to be equivalent. Provide a boron surveillance program wherein the probability of developing a leak in any of the boron envelopes in any of the racks is conservatively accounted for.

RESPONSE

Sufficient prototypical surveillance specimens will be provided which will permit inspection of both faulted (i.e., leaking storage cells) and normal (i.e., leak tight cells). Consequently, the surveillance specimens will provide the information necessary to describe poison material performance in all usage environments.

ENVIRONMENTAL EVALUATION BRANCH

REQUEST NO. 1

Provide the data to support your statement in the November 22, 1978 submittal that realistic assumptions were used to determine that the increment in spent fuel pool storage capacity represents a negligible dose burden. Your response should include the expected dose rates from the spent fuel pool water, spent fuel pool elements and any items that may be stored in the pool, the number of workers that will be exposed to this dose rate from all operations associated with fuel handling in the spent fuel area, occupancy factors and the annual occupational man-rem exposure. Based on your response, justify your conclusion that the occupational exposure will add less than 7% to the total annual occupational radiation exposure burden. Also, justify that this exposure burden is as low as reasonably achievable (ALARA).

RESPONSE

The occupational exposure expected for the Donald C. Cook Nuclear Plant spent fuel pit is estimated to be 20 man-rem, as stated in Section 2.5 of our submittal. The data showing the derivation of this estimate include: (1) the expected dose rate to workers during all operations associated with fuel handling in the spent fuel area, (2) the number of workers associated with fuel handling in the spent fuel area and (3) occupancy factors. These data are provided in our response to request No. 2.

The total annual occupational man-rem exposure received at the D. C. Cook Nuclear Plant, as stated in its Annual Operating Report for 1977, is approximately 276 man-rem per year. Based on this, the occupational exposure of 20 man-rem represents less than a 7% increase to the total annual occupational radiation exposure burden.

Justification that this exposure burden is as low as is reasonably achievable (ALARA) is discussed in our response to request No. 3.

REQUEST NO. 2

In accordance with Section 2.5 of your submittal, the occupational exposure expected for the spent fuel pit is estimated to be 20 man-rem. Provide the data showing the derivation of this estimate. The data should include the expected dose rate to workers (including divers, if any) during each phase of the operation and their occupancy times. Include the exposure that will be received from removal and disposal of the present spent fuel pool racks and miscellaneous equipment presently stored in the pool and installation of the new high density racks.

RESPONSE

The occupational exposure associated with the spent fuel pit modification, was estimated using the sequence of events given below:

I. Removal and Installation of the Racks

1. Unbolting the racks
2. Rack removal
3. Cleaning the pool bottom and performing a radiation survey
4. Installation of the new racks
5. Clean-up

II. Fuel Movement

III. Decontamination and Disposal

1. Washing the racks over the pool
2. Decontaminating the racks
3. Rack cutting and packaging

Dose rates measured one foot above the spent fuel pool surface ranged between 1 to 3 mR/hr. For conservatism, we have assumed a dose rate of 2.5 mR/hr for calculations involving rack removal, installation and fuel movement.

Based on the data available from past experiences with this type of work,[(2)] we estimated a dose rate of 10-20 mR/hr from the racks during the decontamination and disposal time.

The estimated total occupational exposure for the spent fuel pit modification is 20 man-rem. The data used for the derivation are provided in Table 1.

TABLE 1

Procedure	Removal and Installation	Fuel Movement	Decontamination and Disposal
Time	-600 hrs- (10 wks at 6 days/wk at 10 hrs/day)	-64.33 hrs- (193 assemblies at 20 min/assembly)	-75 hrs- (50 hrs for decontamination and disposal as semi-whole at 2 hrs/rack - 25 hrs for additional cutting to reduce the volume significantly at 1 hr/rack)
Dose Rate	2.5 mR/hr	2.5 mR/hr	20 mR/hr
Manpower	7 general crew 2 radiation protection workers Total = 9	3 general crew 1 radiation protection worker Total = 4	3 general crew 1 radiation protection worker Total = 4
Exposure	13.5 Man-Rem	.64 Man-Rem	6.0 Man-Rem

(2) This dose rate is based on actual measurements done by Chem-Nuclear Systems, Inc. during disposal work for the Spent Fuel Pool Storage Rack Modification at the R. E. Ginna Commercial Nuclear Power Plant in Rochester, New York.

REQUEST NO. 3

Describe the method you used to determine the techniques for disposal of the present racks (i.e., crating intact racks or cutting and packaging). Present your consideration of costs and disposal volume as well as the exposure received by the alternative disposal methods in determining as low as reasonably achievable (ALARA) exposure to personnel. State how the present racks will be disposed of.

RESPONSE

The present spent fuel racks will be disposed of as low activity solid wastes. It is unlikely that we would dispose of the present racks intact since it precludes the use of standard shipping packages. The two possible methods of disposal are (1) cutting the racks to significantly reduce the disposal volume or (2) disposing of the racks semi-whole. In both cases the racks would be packaged and shipped to a burial facility for disposal as low level radioactive wastes.

We estimated the occupational dose for the proposed increase in spent fuel pool capacity assuming that the present racks would be cut to reduce the disposal volume significantly. Details of the man-rem calculation are given in response to request No. 2.

The volume of stainless steel in the present racks is approximately 360* cubic feet. However, due to the structure of the racks they occupy approximately 21,050 cubic feet of space. Thus cutting up the racks can result in a significant reduction in the volume of packages that are disposed. We estimate that of the 75 hours projected in our total dose estimate for decontamination and disposal, 25 hours are required for additional cutting of the racks to significantly reduce the disposal volume. Assuming 20 mR/hr and a four man crew, the additional dose for this method is estimated as 2 man-rem.

However, it is our intention to measure the dose associated with the disposal of the racks as we get ready to perform the task. Then taking into consideration alternative disposal costs and doses, we will make our final decision as to the choice of method.

REQUEST NO. 4

Discuss in detail the impact of the proposed Spent Fuel Pit modification on radioactive gaseous effluents for the plant.

RESPONSE

It has been established in the Safety Evaluation Reports of previous spent fuel modification applications (1, 2) that there has not been any significant leakage of fission products from stored spent fuel. Most failed fuel contains small, pinhole like perforations in the fuel cladding at

*This is the correct value. An incorrect value of 11,475 cu.ft. was given in our first submittal dated November 22, 1978 (AEP:NRC:00105). Therefore, the increase in total waste volume resulting from the shipping of the racks would be less than 1%.

reactor operating conditions of approximately 800^{o}F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that the fuel rod temperature is relatively low, approximately 180^{o}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 cladding. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. The only significant noble gas isotope remaining in the SFP (and attributable to storing additional assemblies for a longer period of time) would be Krypton-85. In-plant measurements indicate no noticeable Kr-85 levels above the surface of the SFP.

Assuming linearity for Krypton-85 generation, we estimated that the assemblies stored in the pool at full capacity (2050 assemblies) would release less than 16 curies of Kr-85 as gaseous effluent. The Donald C. Cook Nuclear Plant operating report for the year 1977 indicates that the total yearly Kr-85 release is 174.45 curies for one Unit operation. Thus, the additional Krypton burden to the environment will represent about 7% of the annual Krypton 85 releases for one Unit.

Reference (1) Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to Modification to the Spent Fuel Pool in the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (Northern States Power Company) Docket Nos. 50-282 and 50-306.

(2) Safety Evaluation for Facility Operating License No. DPR-66
Beaver Valley Power Station Unit No. 1
Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company
Docket No. 50-334

REQUEST NO. 5

Provide the history of water leakage from your spent fuel pool.

RESPONSE

The spent fuel pool leak detection system collects any water leakage from the pool and routes it to the 587' level of the Auxiliary Building. Visual observations can be made to determine any leaks in the pool liner. To date no water leakage has been observed.

REQUEST NO. 6

You stated on page 12 in your submittal dated November 22, 1978 that the actual pool bulk water temperature can be expected to be greater than the FSAR design value of 120^{o}F during normal refuelings when the modified pool is filled. If the actual bulk water temperature is expected to be above the FSAR design value under realistic conditions, discuss when this will occur, for what period of time and the effect of this on releases of radioiodine and tritium from the pool.

RESPONSE

Our submittal dated November 22, 1978, page 12 indicates that the Spent Fuel Pool Cooling System is expected to keep the pool bulk water temperature at or below the design value of 120°F during normal refuelings for the full storage capacity of the modified SFP. The complete thermal analysis was submitted to you January 22, 1979.

REQUEST NO. 7

Provide the failed fuel fraction for each fuel cycle of the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2.

RESPONSE

Appendix 14A of the FSAR provides the calculated reactor coolant equilibrium activity for I-131 assuming a 1% fuel failure fraction.

The average iodine activity observed in the reactor coolant for each cycle of the Donald C. Cook Nuclear Plant was compared to this base to determine an estimate of the failed fuel fraction for each cycle. The following are the results of this comparison:

Unit No. 1	Failed Fuel Fraction
Cycle 1	.006%
Cycle 2	.005%
Cycle 3	<.001%

Unit No. 2	Failed Fuel Fraction
Cycle 1	<.001%

Note: Unit No. 1, Cycle 3 and Unit No. 2, Cycle 1 are the current operating cycles.

REQUEST NO. 8

Provide the volume of the pool demineralizer resin bed and the replaceable pool filter.

RESPONSE

The volume of the pool demineralizer resin bed is .85 cu. meters and each replaceable pool filter has a volume equal to .027 cu. meters/filter.

REQUEST NO. 9

Discuss the instrumentation to indicate the spent fuel pool water level. Include the capability of the instrumentation to alarm and the location of the alarms.

RESPONSE

The normal spent fuel pool water level is at 645 feet. Two instruments, RLA-500 and RLA-501, located at the Spent Fuel Pit area are used to detect any significant change in water level.

Instrument RLA-500 actuates the Spent Fuel Pit System abnormal alarm in both control rooms if a high level exists (5.5" above the normal water level). It also actuates the Spent Fuel Pit low level alarm in the Unit 2 control room if a low level exists (5.5" below normal water level).

Instrument RLA-501 actuates the Spent Fuel Pit low level alarm in the Unit 1 control room if a low level exists (5.5" below normal water level).