Telephone (617) 872-8100
TWX 710-380-7619

YANKEE ATOMIC ELECTRIC COMPANY

1671 Worcester Road, Framingham, Massachusetts 01701

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FYR 82-80

July 28, 1982

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

References: (a) License No. DPR-3 (Docket No. 50-29)
(b) USNRC Letter to YAEC, dated May 12, 1982
(c) NUREG/CR-0956, PNL-3065; Commentary on Spent Fuel Storage at Morris Operation; K. J. Eger and H. E. Zima; July 1979; Figure 4, Page 15

Subject: Additional Information - Spent Fuel Pool Modifications

Dear Sir:

The following information is submitted in response to your questions in Reference (b). This information has been previously discussed with your staff and is enclosed herein for documentation purposes.

1. Personnel exposure while the installation of remaining 1st tier racks and all 2nd tier racks is conservatively estimated at less than 1.25 total person-rem. This assumes (a) an area radiation level of 5 mr/hr, (b) 3 operators performing the installations, and (c) a 3-hour installation for each additional rack from yard to positioned in the pool.

2. The dose rate to the operator on the spent fuel pool bridge, assuming 150 day decayed fuel (days since removal from core) shielded with 5 feet of water, is 0.009 R/HR. The integrated 3 minute dose is 0.45 mR.

<u>Analysis Method</u>

A computer model of the fuel assembly was used to determine the dose rate through five feet of water (i.e., 36 feet total water depth during fuel movement) to the operator on the bridge. A correction was made to the computer results to account for the difference between actual versus predicted results. The fact that the measurements were taken under water

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at four feet from the assembly (i.e., the assembly was greater than four feet below the surface) rather than on the surface with the assembly four feet below, may account for the difference in the results. More scatter and, therefore, a higher dose rate would be expected under water.

The computer model is an in-house code named DIDOS-III. DIDOS is a three-dimensional point-kernel shielding code for cylindrical sources. The code determines the direct-shine radiation dose from gamma radiation emanating from a shield self-attenuating cylindrical source. Receptors may be located either on the source axis or at any lateral position beyond the source radius, including positions beyond the source ends. The list of materials available for the source, container, and shield slabs includes air, water, concrete, aluminum, iron, lead, and uranium oxide. Buildup correction is determined for each differential volume using Broder's material-sequence-dependent formula.

The computer generated results compared favorably with the results presented in Reference (c). To be conservative, a correction factor based on actual measurements was applied to the computer generated results. The results are conservative for the reasons given in the first paragraph (i.e., back scatter in the shield was measured by the probe). In addition, the actual dose rate measurement was taken four feet from the pins, not from the top of the assembly. The measurement point was 27 cm closer to the active fuel than in the actual case (i.e., water depth is measured to the top of the assembly). This 27 cm of water represents 0.8 tenth value layers of water for Co-60.

3. Vacuum clean-up of the spent fuel pool walls was considered only during construction when dewatering of pool sections was required to complete the modification. Vacuum clean-up is not necessary during normal periods when the pool is filled.

4. The spent fuel pool purification system can operate with a filter or a demineralizer in service. With the exception of refueling, the demineralizer is in continuous operation. During refueling when water clarity becomes a problem, a cuno filter is placed in service. Usually under these conditions the specific activity of the spent fuel pool water is due to crud contributions representing 70-90% of the total activity. The soluble fission products, cesium-134 and 137, gradually increase in concentration. The release rate based on recent operational data indicates a value of 1.62 uCi's/min. for cesium-137 or 4E-6 uCi/ml. per day. The decontamination factor (D.F.) for different nuclides is dependent on boron concentration. The D.F. range for cesiums and iodines is 20-80. The higher D.F. is experienced at low boron concentrations.

   Assuming a D.F. of 20 and a normal purification flow of 40 gpm, the release rate from fuel cladding would be equivalent to 1438 uCi's per minute to maintain a concentration of 1E.2 uCi/ml. in the pool.

Spent Fuel Pool Surveillance

Monthly Analysis	Limits
pH/cond.	*
Chloride	$<$ 0.15 ppm
Gamma Isotopic **	
Boron	None
Tritium	None

* Dependent on boron concentration
**Daily when moving fuel

I. Filter Replacement with Ion Exchange Bed

a) Chloride $\geq$ 0.15 ppm
b) Cesium-137 $>$ 1E-2 uCi/ml. with soluble activity $>$ 60% of total

Iodine-131 = 5E-3 uCi/ml.

II. Ion Exchange Bed Replacement

a) DF = 1

5. The off-site consequences of dropping a gate section in the spent fuel pool have been evaluated. Although the extent of damage is difficult to predict, it was conservatively assumed that all 721 fuel assemblies release their fuel gap activity as a result of the accident. Fuel gap activity is assumed to be 10% of total gaseous activity. A decontamination factor of 58.5 was used for inorganic iodine (see Ref. c) released under 14 feet of water (conservative for entire pool contents). Other standard assumptions from REG Guide 1.25 (fuel drop accident) were used in the analysis.

Results indicate that a decay period of 90 days would be sufficient to insure that a gate drop accident (using above stated assumptions) would result in acceptability low off-site doses (less than 100 rem thyroid dose).

We trust this information is satisfactory; however, if you have any questions, please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. A. Kay

J. A. Kay
Senior Engineer - Licensing

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