NORTHEAST UTILITIES
THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

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October 28, 1986
Docket No. 50-336
B12303
A06101

Office of Nuclear Reactor Regulation
Attn: Mr. Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Storage of Consolidated Spent Fuel

In May, 1986,(1) Northeast Nuclear Energy Company (NNECO) submitted to the NRC Staff a request to amend its operating license, No. DPR-65, for Millstone Nuclear Power Station, Unit No. 2, to allow storage of consolidated spent fuel in the Unit No. 2 spent fuel storage pool. As a result of the NRC Staff review of this proposal, the NRC Staff forwarded to NNECO a Request for Additional Information.(2) The purpose of this letter is to provide the NRC Staff the requested information.

Question #1:

Because many fuel pin movements will have been involved in the reconstituted fuel assembly, it is likely activated crud will be released from some fuel pins and adhere to some other pins. Show how an increase in dose rates from such crud will be precluded when consolidated assemblies are returned to storage in the spent fuel pool.

Response:

The potential dose rate consequence from crud redepositing on fuel surfaces is insignificant for three reasons. First, the dose rate from crud is a very small fraction of the overall dose rate relative to the fission products in the fuel.

(1) J.F. Opeka letter to A.C. Thadani, dated May 21, 1986, "Millstone Nuclear Power Station Unit No. 2; Proposed Change to Technical Specifications Storage of Consolidated Fuel."

(2) D.H. Jaffe letter to J. F. Opeka, dated September 22, 1986, "Request for Additional Information Millstone 2 -- Storage of Consolidated Fuel in Spent Fuel Pool.

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Second, since the object of concern is the overall dose rate approximately 40 feet away, it is irrelevant whether crud is transferred from one pin to another. And third, nearly all of the crud released will be filtered during the consolidation process rather than redeposited on other fuel rods. The fuel consolidation system contains an elaborate filtration system to control and contain any dispersion of crud released from the fuel rods or fuel assembly. The filtration system is designed to capture the crud at the point of release so as not to allow the crud to spread to any area of the fuel pool.

The filtration system consists of two individual filtration skids, each with a 250 gpm flow rate, with an in-line strainer at the pump inlet to protect the pump from any large debris. Downstream of the pump are two additional stages of parallel filters to minimize filter usage, thereby holding down the volume of waste products.

The filter system is manifolded to all shrouded locations where any fuel rod motion will occur. This manifold and shrouded system provides downward flow in each station. The filter system is sized to ensure sufficient velocity at the tooling to workstation interface to entrain and filter any released crud before it can be dispersed within the work area.

Question #2:

Describe the methodology that will be used to preclude spent fuel pool water from becoming highly contaminated if pin rupture occurs (e.g., close capture cleanup system with filters appropriately shielded), and the method used for identifying pins that are most likely to rupture.

Response:

The shrouded workstations and positive filtration systems (described in the above response to Question #1) will preclude the spent fuel pool water from becoming highly contaminated if a fuel rod ruptures. The first stage of the filtration system is a strainer element which will capture large particles. The second and third stages are pleated paper elements which can capture particles in the low micron range. Further, each filter system is also vented and connected to the plant off-gas handling system. In the event of a pin rupture, one would only expect the release of insoluble, gaseous nuclides (principally Kr-85). The multiple rod pulling tool is fully shrouded and vented to the plant off-gas system. This feature accounts for the collection and disposal of any released fission gases. At Millstone Unit No. 2, the consolidation system will be operated in the spent fuel pool cask laydown area. The cask laydown area gate will be in place during consolidation operations, such that any uncontrolled contamination will be confined to this work area. The system filters are located on skids adjacent to the bottom of the workstation frame. The filters are therefore covered by the full height of the fuel pool water. Filter change out will be accomplished underwater, thus providing adequate shielding.

Prior to consolidation, all candidate fuel assemblies will be pre-inspected with the consolidation system TV cameras to review the condition of the exterior rows of rods. Any fuel assemblies found to have defects will not be consolidated.

Question #3:

Provide all data (mathematical models, parameters, codes and techniques used) that was used in the calculational model devised to determine the increase in dose rates in the spent fuel pool area due to a buildup of radionuclides in the pool water (page 5-4, Attachment 2).

Response:

Calculations have been performed to determine the incremental dose at certain areas around the spent fuel pool due to the proposed increase in spent fuel pool capacity. The dose was first calculated with the pool filled to its present capacity and then predicted for the pool filled to its proposed capacity. A model has been developed to estimate the pool's activity using fuel information available from ORIGEN computer runs generated for the heat load calculations. The model included the following parameters:

1. Li = Leakage of isotope i into the pool from each cycle, in μCi/sec.
2. f = purification flow constant, in sec^{-1}. The constant can be obtained by dividing the purification flow rate by the pool volume.
3. λ i = decay constant of isotope i, in sec^{-1}.
4. γ = fraction of nuclide in the purification system discharge header (If DF-100, then γ = 0.01. If DF=10, γ = 0.1).
5. t = time subsequent to pool insertion, in sec.
6. Ni = activity of each nuclide in the pool, in μCi.

Using the above parameters, it can be seen that the time rate of change of activity in the pool is given as:

$$\frac{dNi}{dt} = Li - (\lambda_i + f - \gamma F) Ni$$

The leakage rate, Li, may be defined as:

$$Li = D\alpha_i N_{fi}$$

where:

D = Fraction of fuel which leaks (i.e., failed fuel - 1%)
αi = Leak rate coefficient for isotope i.
N_{fi} = curies of isotope i in the fuel.

After the fuel is placed in the spent fuel pool, the curies of an isotope in the fuel is governed only by the decay rate and leakage rate.

$$N_{fi} = N_{ofi}e^{(\lambda i + D\propto i)t}$$

where:

N_{ofi} = the curies of isotope i in the fuel at the time the fuel is placed in the pool.

and therefore,

$$Ni = \frac{D\propto iN_{ofi}e^{-(\lambda i+D\propto i)t}\left[1-e^{-(\lambda i+f-\gamma f)t}\right]}{\lambda i+f-\gamma f}$$

The values of N_{ofi} are available from the ORIGEN runs. The leak rate coefficient of the isotopes were those for full power adjusted for cold fuel conditions. The activities of a number of nuclides were calculated using the equation above, but the only two of significant value were those for cesium-134 and cesium-137. The concentrations of cobalt in the pool water were assumed to be:

Cobalt-58: 4.16×10^{-3} microcuries per milliliter
Cobalt-60: 3.61×10^{-4} microcuries per milliliter

This data was consistent with samples obtained from the spent fuel pool of an existing facility and were chosen in lieu of ORIGEN data, since review of the data from several facilities shows that the concentration of cobalt is not directly dependent on the number of assemblies in the pool. Therefore, it was assumed that the concentration of cobalt defined above remains constant for every fuel cycle off-load introduced into the spent fuel pool.

The pool size used is 25 feet by 42 feet by 38.5 feet. The cleanup flow rate used was 125 gallons per minute, and the decontamination factor of the pool cleanup system was assumed to be 10 ($\gamma = 0.1$) for cobalt and 2 ($\gamma = 0.5$) for cesium.

Question #4:

Provide a table showing expected tritium exposures due to the increased fuel storage capacity as the design maximum spent fuel pool water temperature increases from 120°F to 131°F (page 5-5, Attachment 2).

Response:

The design maximum pool water temperature under normal operation with the increased amount of stored spent fuel increases from 122°F to 131°F. This increase in spent fuel pool temperature will result in a slight increase in the evaporation rate from the spent fuel pool. Because the water evaporating is tritiated, the personnel exposure from airborne tritium in the area of the pool will increase correspondingly.

Several models exist for predicting the rate of evaporation from pools open to the general environmental. However, they all tend to predict evaporation rates for

spent fuel pools which are much higher than those which actually occur. Therefore, when calculating the evaporation rate for the spent fuel pool, the following assumptions were made:

- o Because of high pool sweep flow rates directly above the pool, the yearly average air temperatures and relative humidity in the spent fuel pool building will not change as a result of small increases in the spent fuel pool water temperature.

- o All other factors in the mathematical models used to predict evaporation rates remain unchanged.

With these two assumptions, the new evaporation can be predicted as follows:

$$\text{New Evaporation} = \left[\text{Present Evaporation Rate}\right]\left[\frac{Vn-v}{Vp-v}\right]$$

Vn = Water vapor pressure at the new pool water temperature in inches of Hg.

Vp = Water vapor pressure at the present pool water temperature in inches of Hg.

v = Water vapor pressure in the air in inches of Hg.

The dose commitment from tritium is currently below detectable levels (i.e., zero), and as such, no table of expected tritium exposure exists. The calculated increase of 28% in tritium exposure is based on evaporation at 122°F to 131°F and can only be given in relative terms because of the lack of baseline tritium concentrations. The small increase of 28% will ensure that the dose commitment from tritium remains below detectable levels.

Question #5:

Your proposal (Page 1-5) shows a total of 1277 storage locations including 10 spare cells and 217 cells reserved for a full core offload. What is proposed for the remaining 69 locations?

Response:

There are actually 1346 cell locations in the spent fuel pool. The 69 cell locations in question are blocked cells. These locations will remain as blocked cells to accommodate the 3 out of 4 storage pattern required to support the 5-year decay of intact fuel assemblies. Below is a breakdown of actual cell usage.

10	spare cells
217	cells reserved for full core offload
688	cells to contain consolidated fuel
*362	cells with intact assemblies awaiting 5-year decay
1277	total cells containing fuel

*The 362 cells are broken down as follows:

- 157 cells in Region 1, containing intact assemblies
- 205 cells in Region 2, containing intact assemblies, stored in a 3 out of 4 pattern for an actual total of 274 cells, 69 of which are blocked cells.

This provides for:

1277 - total cells containing fuel
69 - cells in Region 2 that are blocked cells
1346 - total cell locations.

We trust you find the above information responsive to your request.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J.F. Opeka
Senior Vice President