OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

SAFETY EVALUATION REPORT

RELATED TO A REQUEST TO REVISE AUTHORITY TO DISPOSE OF

CONTAMINATED DEMOLITION DEBRIS PURSUANT TO 10 CFR 20.2002

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

DOCKET NO. 50-213

1.0 BACKGROUND

Connecticut Yankee Power Company (CYAPCO) has proposed to dispose of 45.5 million kg (100 million lbs) of demolition debris, which includes concrete, concrete reinforcing bar (hereafter “rebar”), some soil, and miscellaneous debris at the U.S. Ecology Idaho Facility. This demolition debris contains relatively low levels of residual radioactive material. Soil above the site Derived Concentration Guideline Levels (DCGLs) and other debris above the levels justified in this proposal will continue to be disposed per 10 CFR 20.2001 at a 10 CFR Part 61 licensed waste facility.

The U.S. Ecology Idaho facility is a Subtitle C Resource Conservation and Recovery Act (RCRA) hazardous waste disposal facility permitted by the State of Idaho. It is located near Grand View, Idaho in the Owyhee Desert. The current cell, that may receive the CYAPCO material, has a capacity of 1.5 million m^3 (2 million yd^3). The most important natural site features that limit the transport of radioactive material are the low precipitation rate [i.e., 18.4 cm/y (7.4 in. per year)] and the long vertical distance to groundwater (i.e., 61-meter (203-ft) thick unsaturated zone below the disposal zone). As is usual with a Subtitle C RCRA site, a number of engineered features are present to enhance confinement of contaminants over the long-term. These include an engineered cover, liners and leachate monitoring systems. Operations at the site include a number of systems that minimize the potential for exposure of workers to any waste handled by the facility. These include a closed facility with filtered ventilation exhaust for transfer of incoming waste material from the shipping conveyance, mechanized equipment for disposition of waste material in the cell, and an application of an asphaltic spray at the end of each day’s operations. The site is permitted to receive non-Atomic Energy Act material or exempted radioactive material that meet site permit requirements.

2.0 TECHNICAL EVALUATION

2.1 SOURCE TERM

As stated in the proposal, “[t]he waste material (the demolition debris) intended for disposal includes flooring materials, concrete, rebar, roofing materials, structural steel, soils associated with digging up foundations, and concrete and/or pavement or

other similar solid materials." The material will be disposed after remediation to remove any areas of high contamination. As the material is demolition debris, it will be in various physical sizes, ranging from the size of sand grains to volumes of several cubic feet. This material will come from portions of the following buildings: (1) containment; (2) auxiliary building; (3) waste disposal building; (4) fuel building; (5) service building; and (6) other miscellaneous structures, soil and asphalt. A total of approximately 45.5 million kg (100 million lbs) is expected to be disposed at the US Ecology Idaho facility. This corresponds to approximately 30,500 m^3 (40,000 yds^3). For dose calculational purposes, CYAPCO assumed that all the waste would be shipped in one year.

CYAPCO has characterized a majority of the structures. The proposal contains a description of the potential source term, focusing on the weighted average characteristics of the total disposal. The weighted average concentration of the waste material is shown in Table 1.

In areas of the spent fuel pool building where characterization has not yet been completed, CYAPCO used the radionuclide concentrations from the Residual Heat Exchanger (RHR) Pit walls for conservatism, which have higher levels of contamination than is expected in the spent fuel building. CYAPCO indicated, in its March 1, 2005, submittal, that it would characterize the spent fuel building after all the spent fuel was removed from the building. However, no characterization plan was provided. In its March 29, 2005 submittal, CYAPCO has committed to taking a minimum of 12 concrete core samples, 20 percent of which will be tested for all 20 radionuclides listed in their License Termination Plan. The U.S. Nuclear Regulatory Commission (NRC) staff finds this commitment to characterize the spent fuel building sufficient for determining the source term for the dose calculations.

Regarding the spent fuel building characterization, CYAPCO also indicated, in its March 1, 2005, submittal, that if results of the (future) characterization indicated concentrations higher than assumed in advance, CYAPCO would inform NRC of the effect of the difference on the conclusions of the 10 CFR 20.2002 request. This approach is unacceptable since §20.2002(a) requires submittal of a description of the waste to be disposed—NRC staff can only review (and approve, if acceptable) what is proposed and described. In its March 29, 2005, submittal, CYAPCO has revised this, to commit to asking NRC to review and approve the effect of potential higher concentrations on the conclusions of the §20.2002 request. In addition, if the results are below the values assumed, CYAPCO will submit the results to the NRC for information. NRC staff finds this change acceptable.

Table 1. Average of Radionuclides in 20.2002 Waste

Radionuclide	Weighted Average Concentration (pCi/g)*
H-3	2.6e+02
C-14	9.7e+00
Mn-54	1.7e-03

Fe-55	1.4e-01
Co-60	2.8e-01
Ni-63	1.7e+00
Sr-90	3.0e-2 (per Sup 3)
Nb-94	1.3e-03
Tc-99	6.5e-03
Ag-108m	2.0e-03
Cs-134	4.9e-03
Cs-137	9.7e-01
Eu-152	5.0e-03
Eu-154	3.8e-03
Eu-155	3.9e-03
Pu-238	3.7e-03
Pu-239	1.2e-03
Pu-241	5.1e-02
Am-241	6.6e-03
Cm-243	1.1e-03

* To convert to Bq/kg, multiply by 37.

The staff finds the surrogate approach originally used (in September 16, 2004, submittal) by the licensee to be appropriate for most radionuclides, but insufficient for C-14. The NRC staff notes that for the resident farmer intrusion scenario, the critical radionuclide is C-14 (contributing about 99% of the dose). Based on the licensee's data (in their Table 8), about 99% of the total inventory of C-14 expected is from material of the containment floor and walls. These materials were described in section 3.3.2 and Table 3 of the proposal. The data in the original Table 3 (September 16, 2004, submittal) included a very wide range of measured C-14 concentrations, as well as a wide range of ratios of C-14 to Co-60. The NRC staff was concerned that there was insufficient data to conclude that there is a consistent ratio of C-14 to Co-60. In addition, the licensee's original proposal stated that the average sample results for C-14 will be applied until additional characterization data is obtained. To address these concerns, the licensee submitted additional characterization data in supplemental submittals of March 1 and March 29, 2005. With the additional data, the licensee also changed its method of evaluating concentrations of C-14, to no longer use the Co-60 concentrations as a surrogate for C-14. The concentrations of C-14 that the licensee used for dose assessment are now based directly on the measured concentrations. NRC staff noted that the averaging calculations performed by CYAPCO for C-14 in the containment building materials included some errors,

related to the dilution of the contaminated layer (only a certain thickness) over the complete thickness of concrete. These errors were corrected by CYAPCO in its March 29, 2005, submittal. The NRC staff concludes that the additional data and averaging methodology for C-14 concentrations are sufficient for the source term for the dose calculations.

The use of the weighted average for estimating dose is reasonable and more realistic than assuming all the waste is at the maximum concentrations.

In its original submittal, CYAPCO did not specify concentration limits or action levels to be applied to individual submittals. CYAPCO provided, in its December 17, 2004, submittal, proposed action levels for individual shipping containers. The maximum allowed concentration in a single shipment will be approximately equivalent to 1.1 Bq/g (30 pCi/g) of Cs-137. The licensee is proposing action levels to limit the radionuclide concentrations in any single shipment. An action level for intermodal containers was proposed to be a maximum exposure rate of 4 FR/hr. Based on modeling using Microshield, that exposure rate would be approximately equivalent to 1.1 Bq/g (30 pCi/g) of Cs-137. A more realistic maximum concentration, as it would also include stronger gamma-emitters like Co-60, would be far lower. The December 17, 2004, submittal indicated that materials might be shipped by intermodal containers or using B-25 boxes. However, the licensee did not provide an action level to be used for shipments with B-25 boxes. To correct this, the licensee provided, in its March 1, 2005, submittal, survey action levels for B-25 containers, based on modeling using the same methodology as was used for intermodal containers. For full B-25 containers, the proposed action level is an exposure rate of 2 FR/hr at 1 meter from the container. For boxes less than full, CYAPCO provided a graph of dose rate versus percentage full to be used. If containers exceed the action levels, CYAPCO plans further characterization to determine if the radionuclide concentrations are acceptable for transfer to the U.S. Ecology site. NRC staff concludes that these action levels are acceptable for preventing the transfer of single containers with excessively high concentrations and exposure rates. For overall dose limitation, the important factor is the average concentration of radionuclides in the entire volume of material to be shipped. Thus, NRC staff concludes that the proposed action levels are adequate.

2.2 SCENARIOS AND PATHWAYS

The licensee evaluated three scenarios for this proposal: (1) transportation worker/driver, (2) disposal facility worker, and (3) resident farmer intrusion scenario. These three scenarios cover the potential release modes for normal operations and the unlikely event of an intrusion after site closure. The intrusion scenario ignores the chemical hazards from intruding on a RCRA disposal facility.

For both the transportation worker/driver and disposal facility worker, the primary pathway is external exposure. The transportation driver is not involved in loading or unloading activities and, therefore, would have minimal potential exposure to inhalation or ingestion hazards. The RCRA site is designed to accept many wastes that are inhalation and ingestion hazards with minimal risk to workers. In the radiation area, the site can receive other naturally occurring radioactive materials (NORM) like uranium and thorium that is not licensed under the Atomic Energy Act. These materials could result in a much higher potential dose exposure through inhalation and ingestion than the radionuclides in the CYAPCO waste. The handling

procedures would be the same for either the NORM or the CYAPCO waste, thereby minimizing the ingestion and inhalation hazard.

The resident farmer intrusion scenario is the bounding scenario for the range of potential exposure pathways. It includes the external exposure pathways, ingestion of food stuff including meat and milk contaminated by irrigation water, drinking water, and inhalation of soil contaminated by irrigation water. The scenario does not include potential intrusion directly into the waste material, because of the depth of final cover and because it is very unlikely that the CYAPCO waste would be in the upper waste layer.

The staff finds these scenarios to be adequate and reasonable for the assessments required for compliance with 10 CFR 20.2002.

2.3 COMPUTER MODELS

The licensee used a combination of Microshield version 5.01 and RESRAD version 6.21 to analyze the three scenarios. The staff finds the use of the codes to be acceptable and reasonable for the conceptual models assessed for this proposal. The staff notes that the use of RESRAD for the intruder scenario is likely to greatly overestimate the dose from the groundwater pathway due to the rather simple conceptual model used in RESRAD as compared to the actual site's likely performance with an engineered cover and a relatively thick unsaturated zone.

2.4 PARAMETER SELECTION

The licensee performed deterministic analyses for the compliance calculations. As such, the parameters used were point estimates. For the Microshield calculations, the assumptions were either appropriate (i.e., dimensions of the vehicle) or bounding (i.e., 1000 hours of occupation for the truck driver/site operator). For the RESRAD calculation, the licensee used a combination of site-specific information and default data sets. The use of site-specific information is the preferred approach, but the use of the RESRAD default deterministic data sets, without adequate justification, is inappropriate, as noted in NUREG-1757, Volume 2. However, the staff performed independent analyses which showed that the impact of the use of these default parameters was small. (See Sections 2.7 and 3.0)

2.5 SENSITIVITY/UNCERTAINTY CALCULATIONS

Since the licensee performed deterministic analyses, the licensee did not supply any sensitivity or uncertainty calculations for review. These are not needed if the deterministic analyses are reasonably bounding.

2.6 LICENSEE RESULTS

The licensee calculated the total dose for each scenario from all radionuclides assuming the average concentration was consistent with the survey data average. For the transportation worker/driver scenario, the licensee calculated a dose of less than 0.01 mSv/yr (1 mrem/yr). For the disposal worker, the dose was calculated to be approximately 0.0145 mSv/yr

(1.45 mrem/yr). For the residential scenario, the dose was calculated to be approximately 0.030 mSv/yr (3.0 mrem/yr).
For a single truck, the licensee has set the maximum dose rate at 4 FR/hr at 1 meter. Assuming an unrealistic bounding scenario where a worker spends half of his or her work year around trucks at the limit (1000 hours) of exposure, this would result in a worker or driver getting 0.04 mSv (4 mrem).

These analyses show that the dose does not exceed a few mrem per year, which the NRC staff finds acceptable for 10 CFR 20.2002 requests.

2.7 INDEPENDENT ANALYSES

The staff, in its review, performed two independent analyses of the potential impact from the waste. The first analysis is a probabilistic analysis of the resident farmer scenario. The second is a comparison using the generic analyses for concrete in NUREG-1640. The results from these analyses allowed the staff to determine that a request for additional information on the data sets used by CYAPCO was not necessary, and provided additional confidence for the staff making the regulatory decision to approve the request.

The first independent analysis used RESRAD version 6.22 in probabilistic mode. Site specific information on the disposal site was used to supplement the probabilistic data set. Correspondingly, the parameter ranges for parameters that had site-specific data were removed from the analysis. The results of the analysis is a dose comparable to the CYAPCO results. The majority of the dose is caused by the C-14 in the waste. Based on past experience with more complex models of disposal cell performance, it is very likely that the RESRAD code is grossly overestimating the release from waste cells to the groundwater. As the results of the licensee’s analysis and the staff’s analysis were similar, no additional information or justification of parameters was considered warranted.

NUREG-1640, “Radiological Assessments for Clearance of Materials from Nuclear Facilities,” is a good reference material of generic assessments of potential doses from disposing of low levels of radioactive materials. For this proposal, the staff used the dose coefficients in Appendix I. The resulting total doses for each scenario resulted a dose that was a small fraction of 0.01 mSv/y (1 mrem/y).

3.0 CONCLUSIONS

The staff finds the dose assessment for the licensee’s proposal to be adequate and reasonable to demonstrate that the potential dose does not exceed a few mrem per year, which the NRC staff finds acceptable. The licensee used appropriate scenarios and computer models. The licensee used appropriate site-specific information. While the licensee failed to justify the generic parameters used in the models, independent staff analyses found the impact of these parameters to be small. As the licensee has used a dose assessment approach using characterization data to estimate the doses, the licensee should keep adequate records, including any additional survey data collected on areas before demolition, to verify the average concentrations. In addition, the licensee has committed to an upper limit on the gamma dose rate from individual trucks.

Based on the above analyses, this material authorized for disposal poses no danger to public health and safety, does not involve information or activities that could potentially impact the common defense and security of the United States, and it is in the public interest to dispose of wastes in a controlled environment such as that provided by the licensed, state-regulated landfills. Therefore, to the extent that this material authorized for disposal in this 20.2002 authorization is otherwise licensable, the staff concludes that the material is exempt from further Atomic Energy Act (AEA) and NRC licensing requirements.

Docket No.: 050-213
License No.: DPR-061

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