CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

JAN 4 2005

CY-05-002
Docket No. 50-213

RE: 10 CFR 20.2002

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D C 20555

Haddam Neck Plant
Request for Approval of Proposed Procedures
in accordance with 10 CFR 20.2002

Connecticut Yankee Atomic Power Company (CYAPCO) proposes to transfer certain of its solid waste from decommissioning of the Haddam Neck Plant (HNP) facilities (e.g., buildings, debris, pavings, soil, etc) to the Waste Control Specialists, LLC (WCS) Facility, located in Andrews, Texas. The purpose of this letter is to request NRC approval of proposed procedures for disposal of certain demolition debris in accordance with the provisions of 10 CFR 20.2002.

A description of the waste material for disposal that potentially contains licensed material is provided in Attachment 1. This description includes the physical and chemical properties important to risk evaluation and the proposed manner and conditions of waste disposal. In addition, CYAPCO has performed a conservative radiological assessment of the demolition debris material and determined that the potential dose to workers involved in the transportation and placement of the waste at the site and to members of the public after closure of the facility as a consequence of the proposed waste disposal will be no more than a few millirem per year Total Effective Dose Equivalent (TEDE) and an insignificant fraction of NRC limits for exposure to members of the public of 25 millirem/yr TEDE(Post Closure).

CYAPCO hereby requests review and approval of this request by March 31, 2005 to support our decommissioning activities at the HNP.

There are no regulatory commitments contained in this letter.

NMSS01

If you should have any questions regarding this submittal, please contact Mr. G. P. van Noordennen at (860)-267-3938.

Sincerely,

Gary H Bouchard 1-4-05

G. H. Bouchard Date
Director, Nuclear Safety/Regulatory Affairs

Attachments:

Attachment 1 Haddam Neck Plant, Evaluation in Support of Alternate Waste Disposal Procedures in Accordance with 10 CFR 20.2002
Attachment 2 Haddam Neck Plant Non-Radiation Worker Dose Calculations
Attachment 3 Haddam Neck Plant Resident/Farmer Dose Assessment

cc: S. J. Collins, NRC Region 1 Administrator
T. B. Smith, NRC Project Manager, Haddam Neck Plant
R. R. Bellamy, Chief, Decommissioning and Laboratory Branch, NRC Region1
E. L. Wilds, Jr., Director, CT DEP Monitoring and Radiation Division

Docket No. 50-213
CY-05-002

Attachment 1

Haddam Neck Plant

Evaluation in Support of Alternate Waste Disposal Procedures

In Accordance with 10 CFR 20.2002

January 2005

Haddam Neck Plant
Evaluation in Support of Alternate Waste Disposal Procedures
In Accordance with 10 CFR 20.2002

1. INTRODUCTION

Approval of the proposed disposal procedures in accordance with the provisions of 10 CFR 20.2002 would allow Connecticut Yankee Atomic Power Company (CYAPCO) to dispose of demolition debris from the Haddam Neck Plant (HNP) decommissioning activities at the Waste Control Specialists Facility in Andrews, Texas (hereafter called WCS site). This attachment provides a conservative assessment of the radiological impacts of the proposed disposal. The following Sections describe disposal site characteristics, the waste material, the radiological assessment and conclusions. The main conclusion is that the potential dose to workers involved in the transportation and placement of the waste at the site and to members of the public after closure of the facility as a consequence of the proposed waste disposal will be no more than a few millirem per year Total Effective Dose Equivalent (TEDE) and a small fraction of NRC limits for exposure to members of the public of 25 millirem/yr TEDE.

It should be noted that this submittal, for conservatism assumes that 100 million pounds of demolition debris with very low concentrations of radioactivity from the Haddam Neck Plant is disposed of at the WCS site. This is the maximum quantity of this type of material. Some of this material may be disposed of at other approved suitable facilities. By using the maximum quantity in this submittal, radiation exposures to the transport workers transporting waste to the WCS site, the workers disposing of the waste at the WCS site and post-site closure members of the public will be maximized.

2. DISPOSAL SITE CHARACTERISTICS

This section describes the features of the disposal facility that are important for the radiological assessment. It describes in turn the geographical and physical environment of the facility, the engineered features, the permits under which the site operates, site operations, radiation monitoring, and post-closure plans. A complete description of the site is provided in documents submitted by WCS to the US NRC in support of an application for a license to authorize near-surface land disposal of low-level radioactive waste (Proposed Radioactive License Number RW-4100). A description of the key features in detail sufficient to support radiological analysis is provided herein.

2.1 ENVIRONMENT AND FACILITY DESIGN

The WCS site is located near Andrews, Texas on the Texas and New Mexico border. Andrews is approximately 77 miles northwest of Midland, Texas. The disposal site address is 9998 West Highway 176, Andrews, Texas 79714.

The most significant natural site features that appear to limit the transport of radioactive material are the low precipitation rate and the long vertical distance to groundwater. The precipitation rate in this arid location is 0.355 meters per year (Reference 7.2 WCS Radiological Environmental Monitoring Summary Report For 2002,). The depth to groundwater accommodates a 5-meter thick cover, a 22.86 -meter thick disposal zone, and a 300-meter thick unsaturated zone between the base of the disposal cell and groundwater (Reference 7.3 Permit No. HW-50358).

A number of engineered features designed to enhance confinement performance have been incorporated in the facility. The most important from the standpoint of radioactive material confinement is the 5-meter thick, low permeability, erosion resistant cover to be constructed at cell closure. This final cover is to be constructed of compacted red bed clay in conjunction with a 40-mil HDPE liner. The HDPE cover liner is to be integrated with a similar liner along the sides and bottom of the cell. The confinement effectiveness of the HDPE liner is ignored in this analysis to assure that projections of potential radiation dose are conservatively maximal.

Together, the low precipitation rate, the thick, low-permeability cover, and the thick unsaturated zone minimize the potential for long term infiltration, dissolution, and transport of constituents to groundwater. The thick cover also minimizes the potential for exposure of waste material radionuclides by erosion or intrusion and minimizes release of radon gas to the atmosphere (although the dose due to the release of radon is shown to be insignificant in these analyses).

Other facility design features and operating procedures provide shorter term confinement of radioactive materials and limit the potential for radiation exposure during receipt of material and emplacement of materials in the cell. WCS adheres to ALARA principles in its Radiation Safety Program (Reference 7.6 WCS Radiation Safety Program).

The total capacity of the cell which could receive the HNP waste is approximately 127,426 cubic meters. The surface area of the cell is approximately 5,574 square meters. The material that CYAPCO proposes for disposal if occupying the full depth of this cell would have a surface area of approximately 1,323 square meters. This means that the HNP material

could occupy approximately 24% of the total volume of this disposal cell. WCS also has two more available cells of the same size which could be used if necessary.

2.2 PERMITS

The WCS site is a Subtitle C RCRA hazardous waste disposal facility permitted under the Texas Administrative Code (TAC) RCRA and TSCA. WCS holds a radioactive material license issued by the State of Texas Department of Health (TDH). In accordance with its regulations and permit conditions, the site has been receiving certain radioactive materials exempt from Nuclear Regulatory Commission licensing requirements for disposal, including material from Honeywell, Mallickrodt Chemical, Molycorp and US EPA Region IV since 2001.

Disposal of radioactive materials at the WCS site is regulated under the State of Texas, TDH and Texas Commission of Environmental Quality (TCEQ). These regulations establish radiation protection standards and permit conditions for disposal of these materials at a permitted disposal facility under the authority of 25 Texas Administrative Code 289.201, "General Provisions for Radioactive Material."

Under the State of Texas TDH general protection standards, all owners and operators disposing of radioactive materials are required to conduct operations in a manner consistent with 25 Texas Administrative Code 289.202, "Standards for Protection Against Radiation from Radioactive Material." In addition, no owner or operator may operate in a manner such that any member of the public would receive an annual TEDE in excess of 100 millirem per year. Additionally, no person may release radioactive material for unrestricted use in such a manner that the reasonable maximally exposed individual would receive an annual TEDE greater than 10 millirem per year.

The facility owner or operator is also required to comply with each of the following license conditions:

- Waste acceptance criteria for radioactive material;
- An environmental monitoring program that monitors air, ground water, surface water and soil for radionuclides and ambient radiation levels in the environs of the facility, and which demonstrates that no member of the general public is likely to exceed a radiation dose of 100 millirem per year from operations conducted at the site.

As previously mentioned, the analysis to follow will show that the HNP material proposed for disposal at the WCS facility will

result in doses that are a small fraction of the applicable limits.

2.3 OPERATIONS

The WCS site accepts only wastes that conform to a documented waste acceptance criteria. This is implemented in the form of a two-step pre-acceptance protocol. In the first step, the generator prepares a chemical and physical characterization of the waste stream on a WCS standard form. The second step is an evaluation performed by WCS to determine the acceptability of the waste. No waste is shipped until the waste is determined to be acceptable by WCS.

Waste acceptance criteria applicable to the material intended for disposal are contained in the following documents.

1. WCS Waste Acceptance Criteria (Reference 7.4)
2. WCS Waste Analysis Plan (Reference 7.5)

WCS is required by condition of its license to operate in a way that assures that the highest potential dose to a member of the public is 100 millirem TEDE per year from operations or 10 millirem TEDE per year from release of radioactive materials for unrestricted use.

To meet these requirements, WCS conducts its operations in accordance with its Radiation Safety Program (Reference 7.6) and other operating procedures. These procedures include measures for minimizing release of material in receipt and handling. Workers use mechanized equipment to transfer and deposit material in the disposal cell. Dust suppression techniques are used daily for materials placed in the cell to minimize the potential for release of radioactive materials to the atmosphere.

To assist in demonstrating compliance with these requirements, WCS also operates a radiation monitoring program. The program includes:

- Personnel dosimetry and bioassay program,
- Periodic collection of grab air samples collected at selected locations in and around the site with analysis for radon progeny, beta and gamma radionuclides,
- Periodic samples of any liquid effluent from within contaminated areas prior to release to offsite bodies, such as sanitary or storm drains,
- Periodic deployment and collection and analysis of passive track-etch detectors with analysis for radon concentration, and
- Periodic deployment and collection of passive dosimeters at locations around the perimeter of the cell with analysis for direct radiation exposure.

The following samples are analyzed for isotopic uranium and thorium, Ra-226, gamma isotopic, gross alpha and gross beta radioactivity:

- Periodic collection of grab air samples during material transfer operations,
- Periodic collection of continuous air samples from the admin/lab area,
- Periodic collection of soil samples from locations downwind of the disposal area, and
- Periodic collection of groundwater samples from 18 monitoring wells (8 up gradient and 10 down gradient) with analysis for gross activity.

2.4 POST-CLOSURE PLAN

As required by the State of Texas, TCEQ, WCS maintains an approved closure plan, submitted as part of its permit application (Reference 7.3). The plan conforms to all standard closure and post-closure requirements applicable to RCRA disposal facilities, including post-closure monitoring and financial assurance.

The plan provides reasonable assurance that the general radiation protection standard for the public (TEDE of 10 millirem per year) will not be exceeded. It should be noted that this standard for post closure exposure to a member of the public is set below the NRC standard for unconditional release of an NRC licensed facility which is 25 millirem per year TEDE.

3. DESCRIPTION OF WASTE

3.1 Physical Properties

The waste material (i.e., 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. Soils remediated for the purpose of meeting the final status survey requirements of the HNP License Termination Plan (LTP) (i.e., soils that exceed the Derived Concentration Guideline Levels (DCGL) in the LTP) will not be disposed of at the WCS facility as the concentrations of the key gamma radionuclides at the DCGL levels are approximately an order of magnitude over the averages determined later in this evaluation.

The demolition debris proposed for disposal at the WCS facility will originate from the demolition and removal of structures and paved surfaces at the HNP plant site, after they have been decontaminated to remove areas that are highly contaminated.

The physical form of this demolition debris will be that of bulk material of various sizes ranging from the size of sand grains up to occasional monoliths with a volume of several cubic feet. CYAPCO, for the purpose of calculations, assumed the material to be a homogeneous mixture with a specific density of 1 gram per cubic centimeter during shipment and 1.50 grams per cubic centimeter after compaction in the disposal cell at WCS. The material will be dry solid waste containing no absorbents or chelating agents.

3.2 Estimated Waste Volume

It is estimated that the mass of demolition debris originating from the decommissioning of the HNP will total approximately 100 million pounds. A breakdown of this waste by source is shown in Table 1. With an assumed density of 1.50 grams per cubic centimeter, (after compaction at the disposal site) the estimated volume of material to be disposed of at the WCS facility is approximately 40,000 cubic yards. This represents approximately 6 percent of the annual volume of waste that the WCS facility receives for landfill purposes. This volume of waste will not place a burden on the operations of WCS as it is anticipated that waste will be shipped to the WCS facility throughout 2005 and 2006. This amounts to approximately 1,250 intermodal shipments per year.

The material will not be isolated or dedicated to a single burial cell at the WCS facility. Rather, it will be co-mingled with other radioactive and non-radioactive waste material. Dust suppression techniques are used daily for materials placed in the cell to minimize the potential for release of radioactive materials to the atmosphere.

3.3. Radiological Characterization of Waste

3.3.1 Background:

CYAPCO has been in the process of characterizing the radiologically contaminated buildings on site. Some radiological data is available on all buildings in the radiological controlled area. The demolition plans are to scabble off surface concrete where contamination levels are high and to dispose of this material at radioactive waste disposal facilities other than the WCS facility. Areas of concrete where high neutron flux has caused significant activation of the concrete are also not proposed for disposal at the WCS facility. After dispositioning the surface contaminated material containing the highest levels of radioactivity, the remainder of the building and structures will be demolished and it is proposed that much of the debris be shipped to the WCS facility near Andrews, Texas. For the purpose of determining the radioactivity level of material to be shipped to the

WCS facility, concrete core sampling is most appropriate as these portions of the applicable buildings will be demolished in total. The demolition process results in mixing the surface and volumetric contamination with the remainder of the wall and floor material. This makes the average concentration in the total thickness of the wall or floor appropriate in determining the overall radioactivity content of the waste material. Additional sampling will be conducted during building demolition to confirm radionuclide waste concentrations and scaling factors where currently available information is limited. It is also appropriate to use average values as the dose limits are in terms of annual exposures. Any variation of the waste shipments would be incorporated in the average of all shipments made during a year.

Structural materials other than concrete are expected to have only low levels of surface contamination and are therefore bounded by the characteristics of the concrete intended for disposal. Any rebar encased in concrete is also expected to have much less than the surface contamination levels as it is located below the depth to which most of the surface contamination is located and therefore can be treated the same as the concrete.

3.3.2 Characterization Results

The portions of site buildings (including structural material after removal of contaminated system piping and components) that CYAPCO proposes to dispose of at the WCS Facility are as follows:

- Containment Walls (including the containment liner) above elevation 17.5',
- Containment Floors/Structures that are inside the containment liner,
- Residual Heat Exchanger (RHR) Pit (a portion of the Primary Auxiliary Building) Floors,
- RHR Pit Walls,
- Waste Disposal Building Floors,
- Waste Disposal Building Walls and Ceilings,
- Remainder of the Primary Auxiliary Building above the RHR Pit,
- Spent Fuel Pool Walls and Floor, (after liner is removed)
- Remainder of the Fuel Building above elevation 17.5',
- Service Building above elevation 17.5', and
- Other Miscellaneous Radiological Controlled Area (RCA) Structures, Soil and Asphalt.

A breakdown of the estimated quantities of materials from the above sources is included in Table 1. The following discussion describes the operational history of the buildings that will make up the waste to be disposed of at the WCS Facility and characterization results for

the waste that will result from their demolition. In this analysis, some conservatism is applied where data gaps exist. As previously mentioned, additional sampling will be conducted as part of ongoing decommissioning activities to fill these data gaps.

Containment Walls above elevation 17.5'

This portion of the Containment Building has not experienced high levels of contamination due to its location and the fact that it did not come in contact with contaminated system leakage. The concrete in this area is outside of a steel liner that covers the entire inside wall of the containment dome above elevation 17.5'. The liner itself is not expected to be highly contaminated. Four (4) Concrete Core Bores have been taken from quadrants of the containment wall at approximately elevation 4'. Twenty-Four (24) wafers cut from these cores were analyzed for gamma radionuclides, tritium and selected wafers for hard to detect radionuclides. Concrete at this elevation will not be shipped as waste but provides conservative characterization results. The containment wall at elevation 4' has been potentially exposed to more contamination than higher elevations of the wall. This elevation is below the water table and therefore subject to the diffusion of contaminated groundwater that has been present outside the containment. The inside of containment at this elevation was exposed to standing water during the cavity seal failure event in 1984. Therefore, using the wall characterization results from elevation 4' for elevations above 17.5' is conservative. The results of these characterization samples are contained in Table 2. It should be noted that most of the sample results included in Table 2 indicated no detectable activity at the Minimum Detectable Activity (MDA) concentration. For this reason, the scaling factors determined in Table 4 for the RHR Pit Floors was used to determine waste activities for this building area for all radionuclides except H-3, Co-60, Sr-90 and Cs-137 for which actual sample averages were used. For C-14, the scaling factor to Co-60 from core sample # 181 taken in this area will be used.

Containment Floors and Internal Structures

As described in Section 3.3.1 "Background" above, the highly contaminated surfaces and significantly activated areas of buildings will not be included as waste materials for the WCS facility. Therefore, the core sample results for the in core sump and the main containment sump are not applicable to the determination of average waste concentration for this area of the building. These sample results were used to determine scaling factors for radionuclides not analyzed for all of the samples. The lower area of

the inside of containment experienced standing water from the cavity seal failure previously mentioned. Six (6) cores were taken from the floor in this area. Twelve wafers from these cores were analyzed for gamma and selected hard to detect radionuclides. As can be seen from Table 3, the contamination is one to two orders of magnitude higher in the first 2.5" wafer of the cores compared to the wafers from deeper regions of the cores. Three additional cores from the floor and walls of this area were taken in 1999. These cores showed a similar trend. For Co-60 and Cs-137, the concentration of the resulting debris waste, C_{waste}, was determined by averaging the concentration in the first 21/2 inch wafers, C_1, of the 12 cores and distributing that value over the average thickness of the containment internal floors, X(in), as follows:

$$C_{waste} = C_1 \frac{2.5}{X}$$

These characterization results were also be used for the internal structures of containment that will be disposed of at the WCS facility. Using the floor samples for the internal structures is conservative as most of this material was above the cavity seal failure event and any areas of high surface contamination will be remediated and shipped to a facility other than WCS. Data from areas where CYAPCO has collected both floor and wall samples show the floor samples to be at least an order of magnitude higher in radioactivity content than the wall samples.

Whereas many of the samples did not show any detectable activity for most radionuclides, the average scaling factor calculated from the surface wafers was used to calculate the average activity for all radionuclides except H-3, C-60 and Cs-137. For these radionuclides, the average of the sample results was used to characterize the waste. Two (2) samples from the highly contaminated containment sump were analyzed for tritium. These two sample results were averaged to determine the waste concentration. It is expected that when more wafers are analyzed for H-3, the average concentration will be reduced.

The characterization samples for this area show measurable levels of C-14. This radionuclide has not been detected in concrete outside of the containment liner in the containment wall or in other buildings on site. It is possible that a gaseous diffusion mechanism has resulted in the shallow permeation of C-14 into containment interior concrete. Therefore, the average sample results for C-14 inside the containment liner will be applied for all concrete inside the containment liner until additional characterization data is obtained.

Residual Heat Exchanger (RHR) Pit of the Primary Auxiliary Building Floors

The RHR Pit is one of the most contaminated areas inside buildings at the HNP site due its design. For this reason, it was targeted for concrete characterization work. Four (4) core bores were taken from the floors in this pit. Seventeen wafers were cut from these bores and analyzed for gamma radionuclides. A subset of the wafers was also analyzed for tritium and all other hard to detect radionuclides. Table 4 details the results of the analysis performed on these core bores. As can be seen in Table 4, contamination (with the exception of Tritium) is highest in the first 2.5" wafer and drops by approximately two orders of magnitude for the second and subsequent wafers. As discussed earlier, areas of high contamination will be remediated. For conservatism, it will be assumed that no scabbling will be performed in the areas where the cores were drilled prior to demolition of the RHR Pit floors. The only sump in this building is known to be highly contaminated and has been remediated. None of the cores were taken from sump areas. The determination of the average concentrations for Co-60 and Cs-137 follows the same method as was used for the containment floors above.

To determine the activity of the other radionuclides except tritium, the scaling factors for Co-60 calculated from Sample # 165 taken in this area were used to determine the average concentration. For Tritium the average concentration in all the samples was used to characterize the waste. This is due to the fact that tritium acts as water when diffusing into the concrete.

RHR Pit Walls

Five (5) core bores were taken from the walls of the RHR pit. One was taken through an internal wall at approximately elevation -17'. The other four were taken through the exterior wall of the pit at three subsurface elevations at the location adjacent to the outside location of the former Refueling Water Storage Tank. This was determined to be an area of high potential for elevated concrete contamination due to diffusion of contamination from outside of the building resulting from leakage of the Refueling Water Storage Tank (RWST) while the plant was operating. Twenty-Eight (28) wafers from the cores were analyzed for gamma radionuclides and tritium, and a subset for all other hard to detect radionuclides. The results of these analyses are shown in Table 5. Surface and volumetric radioactivity levels were generally low with the exception of Tritium which was detected in

moderate levels (compared with the Derived Concentration Guidelines Levels published in the HNP LTP). As contamination was seen in both end cores and on certain internal wafers, the average concentration of H-3, Co-60, Sr-90 and Cs-137 from all the cores was used in determining the waste concentration from this building area. This approach is conservative as the highest concentrations are in the outside wafers and samples from only 15 % of the core length were analyzed. By not including results from all interior areas of the cores where lower concentrations are expected, the average concentration is higher and, therefore conservative as the average does not take credit for all the dilution that will occur when the building is demolished.

For the remaining radionuclides there were essentially no detections at the MDA concentration. For this reason and due to the relatively low levels of Co-60 for samples in this area, the scaling factors determined for the RHR Pit Floors were used to determine the waste concentrations for the remaining radionuclides.

Waste Disposal Building Walls, Ceilings and Floors above elevation 15'

The waste disposal building has been decontaminated (this waste was sent to a facility other then the WCS facility) to allow for open air demolition. The building above elevation 15' has been demolished and the waste is currently stored at the site for future disposals. The current plan is to demolish all of the building and dispose of this post decontamination debris as radioactive waste at WCS. Surveys of the building were reviewed to determine the relative contamination level of this building other than the floor areas. The result of this review is that these portions of this building had low levels of contamination, at least an order of magnitude below the levels on the RHR pit floors and somewhat lower than the levels on the RHR Pit walls. A very small percentage of the building areas were contaminated. The concentrations determined for the RHR Pit walls was used for these areas for conservatism.

Waste Disposal Building Floors at Elevation 0'

As mentioned above, the basement floor of the Waste Disposal Building will be removed in its entirety. One concrete core sample was taken from the basement floor in 1999. Three wafers 0.5" thick were cut from the floor side of the core and analyzed for gamma radionuclides. The results for these wafers are shown on Table 6. The results show shallow contamination at levels consistent with the RHR pit floor samples. These results will be used to characterize the

waste from this area along with the scaling factors determined from the RHR Pit floor samples. The tritium sample results from the RHR floors samples were used to characterize this area. A review of surveys of the area shows that the contamination levels are generally low and at least a factor of 5 below the levels on the RHR Pit floors. Only a small portion of these floors are contaminated. Using the results of the one core is therefore conservative.

Primary Auxiliary Building (PAB) other than RHR Pit

All of the Primary Auxiliary Building other than the RHR Pit has been decontaminated to allow open air demolition with the scabbled material disposed of in the manner of other higher contaminated materials (not at the WCS facility). The remaining material will be demolished and the debris is proposed for shipment to the WCS facility. One concrete core was taken from the pipe trench portion of the PAB in 1999. The pipe trench portion of the PAB is one of the most highly contaminated areas in the PAB. A review of contamination levels of other areas of the PAB shows generally low contamination levels with only a small portion of the building posted as a contaminated area. When the average contamination levels in the PAB as whole are considered, the characterization results for Co-60 and Cs-137 for the containment floors are conservative and are applied to all areas of the PAB except for the RHR Pit. As with the other areas outside of the containment liner, the scaling factors for Co-60 from the RHR Pit floors was used to determine the concentrations of radionuclides other then H-3, Co-60 and Cs-137. The average concentration of H-3 for the RHR Pit walls was conservatively used for these upper areas of the Primary Auxiliary Building.

Spent Fuel Pool Walls and Floors below elevation 17.5'

The spent fuel pool in the Fuel Building is lined with a stainless steel liner. The demolition plan for this building is to remove the liner after the pool is empty of fuel and all other material. This liner will be disposed of at a waste facility other than the WCS facility. It is known that at least a small amount of leakage past the liner has occurred. Due to concerns with the integrity of the fuel pool there has been no characterization of this area. The high concentrations measured for the RHR Pit floors will be used to represent this area. It is assumed that all concrete in this area will be sent to the WCS facility after removal of highly contaminated areas.

Remainder of Fuel Building above Elevation 17.5'

The review of surveys of the remainder of the fuel building has shown low contamination levels with only a few small contaminated areas. Although the RHR Pit walls show higher levels of contamination, the concentrations for the RHR Pit walls was used for the remainder of the Fuel Building for conservatism. It is assumed that all concrete in this area will be sent to the WCS facility for disposal.

Service Building above Elevation 17.5'

The Service Building has not experienced many contamination events. A review of building surveys shows only a few small contaminated areas in a Decontamination Room and the Chemistry Lab. Contaminated commodities in these areas will be removed and shipped to facilities other then the WCS facility. The remaining concrete, will on the average, have very low contamination levels. The expected levels are consistent with those in the Containment walls and therefore, those concentrations will be used for the service building with the exception of C-14. As there have been no detections of C-14 in concrete outside of the containment liner, the scaling factor for Co-60 determined for the RHR Pit floors was used to determine the C-14 waste concentration in this area.

Miscellaneous Structures, Soil and Asphalt

There are other relatively small structures which are in the Radiological Controlled Area but have very low contamination levels. These include the Cable Vault and the Radwaste Reduction Facility. It is planned that the portions of these buildings above elevation 17.5' be disposed of at the WCS facility. These buildings have a very low contamination history and either have very small or no contaminated areas. There will also be quantities of slightly contaminated soil that will be displaced to allow access for removal of foundations. Quantities of slightly contaminated asphalt will be removed from the site to meet non-radioactive site closure criteria. As previously discussed, soil with radionuclide concentrations near the LTP DCGLs will not be disposed of at the WCS facility as these levels would be inconsistent with the concentrations in other types of waste proposed for disposal there. Waste concentrations determined for the containment walls are appropriate for application to this class of waste materials except as amended for C-14 as was done for the Service Building.

3.3.3 Average Concentration of Waste to be shipped to WCS

In order to determine the average concentration of waste proposed to be disposed of at WCS, a weighted average of the concentrations discussed earlier is determined in Table 8. It can be seen from characterization sample results in Tables 2 thru 7 that the primary radionuclides that affect dose to personnel either transporting the waste or working with its disposal at WCS are Co-60 and Cs-137. All other gamma emitting radionuclides are present at much lower levels and therefore, need not be included in calculating worker dose. The alpha and beta emitting radionuclides are not a direct dose concern and can only be an inhalation or ingestion hazard during placement in the disposal cell. The controls, discussed earlier, those present at the WCS facility and the relative low concentrations will preclude any significant dose from these radionuclides to the workers.

The weight of waste from each building is shown based on a recent estimate. The concentration of Co-60 and Cs-137 for each building area is multiplied by the estimated weight of building debris from each building, summed and the sum divided by the total waste weight to determine the weighted average. This value is shown in Table 8. These values are used later to determine expected yearly dose due to transportation and WCS site workers involved in disposal of the HNP material.

For the purposes of determining potential dose to a member of the public after the closure of the WCS site, the activities of other radionuclides will be determined by the use of scaling factors based on actual HNP characterization sample data. The sample data indicates different scaling factors for inside versus outside of the containment liner. This is primarily due to the detection of C-14 in concrete inside of the containment liner. Therefore, two sets of scaling factors were used to characterize the waste. As shown on Table 8, one set of scaling factors determined from the average of sample results inside the containment liner will be applied to that area. Scaling factors determined from RHR Pit floor samples will be used for concrete and other materials from outside of the containment liner. All scaling factors will be based on the ratio of the hard to detect radionuclide to Co-60. As previously discussed, average sample results were used to characterize the proposed waste for tritium. Sr-90 samples show some limited degree of migration of this radionuclide through concrete structures in certain plant areas. As can be seen in Table 8, average sample results for Sr-90 were used when this was the case. A review of the sample data shows that the scaling factors determined are conservative as many are based on sample results that indicate no detectable activity

at Minimum Detectable Activity (MDA) concentration rather than actual detections. Using the above outlined protocol, Table 8 illustrates the values used to determine the average waste concentrations for the material proposed for disposal at the WCS facility.

4 RADIOLOGICAL ASSESSMENTS

4.1 Transport Worker Dose Assessment (Attachment 2)

The Transportation Scenario Maximally Exposed Individual (MEI) dose equivalent will not exceed a few (e.g., five (5)) millirem/yr. This standard of a "few mrem/yr" to a member of the public prior to license termination is defined in NRC Regulatory Issue Summary 2004-08 (Reference 7.1). The transportation workers and worker at the WCS facility are treated as members of the public as the WCS site is not licensed by the NRC to receive by-product radioactive material for disposal. Evaluations of both internal and external dose hazards to the transportation worker are discussed below.

Each conveyance will be a strong-tight container and will be verified to be in compliance with Department of Transportation (DOT) external loose surface contamination limits prior to shipment. Therefore, there are no internal dose hazards associated with the Transportation Scenario.

The conservative average activity concentrations discussed in Section 3.3.3 of this Attachment were used to calculate dose to members of the public transporting waste to the WCS facility. Attachment 2 contains a dose assessment using the TSD-DOSE computer code to calculate the dose to a driver transporting a waste shipment from the HNP site (containing debris at the average concentration previously determined) for the 2 hour trip (maximum expected time) to the rail loading station. At the rail loading station, the intermodal box will be loaded onto a rail car for the remainder of the trip to the WCS site. For members of the public involved in transportation of the waste, the truck driver will be the person receiving the highest dose due to the length of time within the vicinity of the shipment.

The dose to the driver from a box containing building debris at the average concentration calculated in Section 3.3.3 is 7.2 E-5 mrem. If one driver is conservatively assumed to transport 250 loads (approximately 20% of the HNP's yearly debris shipments with 50% of the total waste being shipped in each year 2005 and 2006) to the rail loading station, the driver would receive 0.019 mrem/yr. Using the upper limit of no more then 5 mrem/yr to a member of the public, the waste in all the shipments could be 267 times the average concentration and still be within the allowable yearly dose.

4.2 Disposal Facility Worker Dose Assessment (Attachment 2)

Attachment 2 also calculates the dose to workers at the WCS disposal site. The two tasks during which the workers are exposed to the HNP waste material are receiving (i.e., weighing/inspecting) the shipment and transporting the intermodals to the landfill and unloading into the disposal cell. Each task will be analyzed separately.

The persons receiving the waste will receive a dose of 9.1 E-5 mrem per shipment of the HNP waste material (at the average concentration determined in Section 3.3.3) received. As discussed in Attachment 2, 10 different workers could be utilized in the receipt and disposal of the HNP waste material. It is, therefore, conservative to assume that one worker will perform the receipt work for half of the shipments received in a year from the HNP waste (625 intermodals), the total dose received would be 0.057 mrem/yr. Using the upper limit of no more then 5 mrem/yr to a member of the public, the waste in all the shipments could be 88 times the average concentration and still be within the allowable yearly dose.

The workers on the WCS site moving the waste from the receiving area and placing it in the disposal are calculated to receive a dose of 1.5 E-4 mrem per shipment of the HNP waste material (at the average concentration determined in Section 3.3.3) received. If it is assumed that the remaining 8 workers at the WCS site dispose of the HNP waste material in teams of 2, they would each be exposed to 25% of the HNP waste material. This would involve handling 313 intermodals and result in a total dose 0.047 mrem/yr. Using the upper limit of no more then 5 mrem/yr to a member of the public, the waste in all the shipments could be 106 times the average concentration and still be within the allowable yearly dose.

4.3 Offsite Individual/Population Dose Assessment (Attachment 2)

The TSD-DOSE code output contained in Attachment 2 also calculates offsite individual and population doses. These results will be analyzed as follows:

Attachment 2 gives a dose to an offsite individual of 4.1 E-9 mrem from a shipment of the HNP waste demolition debris at the average radionuclide concentrations. For all the 1250 shipments in a year, the dose to an offsite individual would be 5.1 E-6 mrem/yr. Using the upper limit of no more then 5 mrem/yr to a member of the public, the waste in all the shipments could be 980,000 times the average concentration and still be within the allowable yearly dose.

The average worker dose can also be confirmed using the worker population dose results in Attachment 2. The total worker population dose for one shipment at the average concentrations is given as 4.6 E-7 person-rem. For all 1250 shipments in a year, the total worker population dose would be 0.575 mrem. As discussed earlier there is expected to be 10 workers involved in the receipt and disposal of the HNP waste material at WCS. The average worker dose would, therefore, be 0.0575 mrem/yr. Using the upper limit of no more then 5 mrem/yr to a member of the public, the waste in all the shipments could be 87 times the average concentration and still be within the allowable yearly dose.

4.4 Maximum Waste Concentration Limit

As can be seen in the attached tables, the samples concentrations vary greatly with some locations with concentrations in the range of 50 to 500 times the average concentration for all the waste. The above dose analysis indicates (using the lowest calculated multiplier) that if all of the waste shipped in a year had a concentration 87 times higher then the average concentration, the maximum allowable dose to an individual would be less then 5 mrem. Considering this and the concrete sample results, CYAPCO intends to set the maximum concentration on any shipment of material to WCS at a value approximately 40 times higher then the average concentration (monitoring of this limit will be discussed in Section 5). Using these limiting values and the dose analysis presented above, the highest dose to any member of the public would be 2.3 mrem/yr (for the receiving worker) if all the waste containers were at the limit. It is expected that few if any shipments will have this maximum concentration and the yearly average dose to drivers and WCS site workers will correspond with the much lower dose from material having the average radioactivity concentrations determined in Section 3.3.3.

4.5 Resident/Farmer Dose Assessment (Attachment 3)

The RESRAD computer code was used to calculate the projected effect of the proposed disposal activity on future residents at the disposal site. Each isotope of concern was included at a soil concentration of one (1) pCi/g, such that the resultant calculated dose equivalent to the maximum exposed individual (Resident Farmer) could be evaluated in terms of mrem/year per pCi/g activity concentration. A comprehensive report describing the methodology, input parameter selection, and calculation results is included as Attachment 3. It can be seen in Attachment 3 that tritium and Pu-238 are the only radionuclides that have a high enough mrem/yr per pCi/g to be listed in the RESRAD results.

The average radionuclide concentrations in the waste proposed for disposal were determined in Section 3.3.3. It can be seen in Attachment 3 that many of the assumptions used in the RESRAD code were conservative resulting

in an over estimation of post closure dose to a member of the public. The calculation of expected dose to a member of the public after closure of the facility is performed as follows: Table 9 shows the dose to the Resident Farmer for each radionuclide at a concentration of 1 pCi/g. Table 9 also shows the post-closure dose to a member of the public when the dose at 1 pCi/yr for each radionuclide is scaled to the average concentrations of the HNP waste determined in Section 3.3.3. It can be seen that the total expected dose to a member of the public post closure is 1.101 E-5 millirem/yr.

The above discussion demonstrates that the disposal of the HNP waste materials at the average isotopic contaminant concentrations described in Section 3 will result in a dose that is approximately six (6) orders of magnitude below the 25 millirem/yr NRC post closure criteria for allowable dose to a member of the public (and also the WCS site standard of 10 millirem/yr general radiation protection standards for the public) and is of little significance when the high factors of conservatism used in the calculation are considered.

5. WASTE ACTIVITY MONITORING SURVEYS AT THE HNP

CYAPCO has performed Microshield© runs to determine an on-site survey limit for the disposition of waste in appropriate containers that can be shipped to WCS disposal site.

An action level has been developed to identify when it is appropriate to transport a container to WCS or to an alternate disposal site should the container dose rates exceed the alternate waste disposal procedure criteria of 10 CFR 20.2002. These action levels are expressed as a dose rate (in µr/hr), and are based upon the assumption that all gamma emissions are produced by the decay of a mixture of Cs-137 and Co-60 that corresponds to that shown in Table 8. The action levels also assume that the contents of the container contains this mixture of Cs-137 and Co-60 contamination at maximum allowable activity concentrations that are 40 times the average values as has been discussed in Section 4.4.

It should be noted that as the weight (and effective density) of the contents of each container is variable, the action levels will also vary. As shipment to WCS will be by intermodal containers, action levels were determined for these containers using appropriate software (i.e., MicroShield).[1]

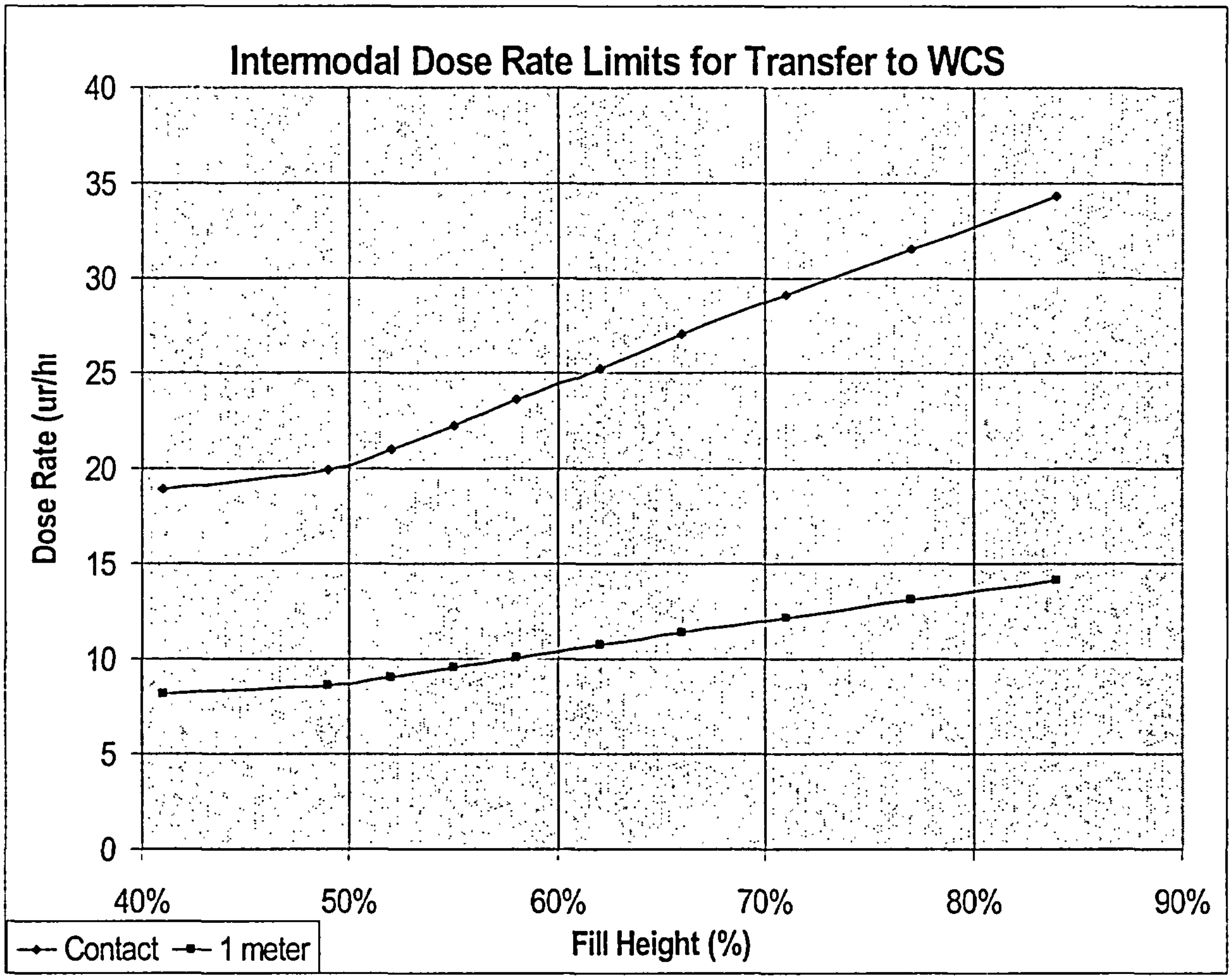
Using a nominal container fill height for an intermodal container corresponding to 55 %, a 1 meter dose rate of 10 µr/hr over background (See Figure 1) is selected as a reliable and conservative action level for determining compliance with the alternate disposal procedure criteria. It is considered that containers exhibiting dose rates below the action level may be shipped to WCS and those exhibiting higher dose

[1] MicroShield 5, Grove Engineering, Rockville, MD, 1998.

rates need to be shipped to alternate facilities or investigated further to determine radionuclide concentrations.

As discussed in Section 4.4 using the maximum allowable survey dose at 1 meter from each intermodal container of 10 uR/hr for all containers would result in a worst case dose to a member of the public of 2.3 mrem/yr. This is well within the criteria of 5 mrem per year established by NRC for this alternative disposal procedure. In conclusion, the use of a 10 uR/h for an intermodal at 1 meter for waste to be disposed of at the WCS site would result in worker exposures well within the NRC criteria for approval of the alternate disposal request in accordance with 10CFR20.2002.

Figure 1



6. Conclusions

Based on the above assessment, it can be concluded that the calculated potential dose to members of the public as a consequence of the proposed waste disposal from the decommissioning activities at the HNP at the Waste Control Specialists, Andrews, Texas Facility are as follows:

- Workers involved in the transportation to and placement of the waste in the disposal cells at WCS will receive doses that are a fraction of the 5 mrem/yr dose allowable for this type of activity.
- The projected dose to residents after closure of the site is an insignificant fraction of the 25 millirem per year limit.

Therefore, CYAPCO concludes that the proposed request for approval in accordance with 10 CFR 20.2002 will not have a significant impact on the workers, public, or the environment and that it is, therefore, acceptable.

7. References

7.1 NRC Regulatory Issue Summary 2004-08, Results of the License Termination Rule Analysis, dated May 28, 2004

7.2 WCS Radiological Environmental Monitoring Summary Report for 2002

7.3 WCS RCRA Permit No. HW-50358

7.4 WCS Waste Acceptance Criteria

7.5 WCS Waste Analysis Plan

7.6 WCS Radiation Safety Program

7.7 TSD-DOSE, A Radiological Dose Assessment Model for Treatment, Storage, and Disposal Facilities, US Department of Energy (DOE)

Table 1

Estimated Waste Quantities Proposed for Disposal at Waste Control Specialists

Source of Waste	Estimated Waste Weight (pounds)
Containment Walls	40,000,000
Containment Floor & Internal Structures	20,000,000
Residual Heat Exchanger(RHR) Pit Floors	1,000,000
RHR Pit Walls	2,000,000
Waste Disposal Building Walls	2,500,000
Waste Disposal Building Floors	500,000
Remainder of Auxiliary Building (w/o RHR Pit)	7,000,000
Spent Fuel Pool Walls & Floor	1,000,000
Remainder of Fuel Building above elevation 17.5'	8,000,000
Service Building above elevation 17.5'	8,000,000
Miscellaneous Structures/Soil/Asphalt	10,000,000
Total	100,000,000

TABLE 2
Containment Wall (Outside of Liner) Samples

	Sample # 181						Sample 182						Sample 183						Sample 184							
Radio-nuclide	181-1C-01	181-1C-02	181-3C-01	181-4C-01	181-6C-01	181-6C-02	182-1C-01	182-1C-02	182-4C	182-5C-01	182-9C-01	182-9C-02	183-1C-01	183-1C-02	183-5C-01	183-6C-01	183-9C-01	183-9C-02	184-1C-01	184-1C-02	184-3C-01	184-5C-01	184-8C-01	184-8C-02	Average of all Containment Wall Samples	Scaling Factor for Containment Walls to Co-60 (Using Sample 181-1C-01 for C-14, RHR Pit Scaling Factors for others)
	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	
H-3	3.85	2.55	2.63	13.20	2.31	4.28	8.37	2.56	2.43	11.40	2.34	2.47	2.31	7.54	5.29	2.48	2.09	2.28	6.28	3.10	13.00	2.39	2.26	24.10	5.48	Use Sample Avg.
C-14	0.51	0.51					0.54	0.51					0.52	0.57					0.56	0.57					0.54	2.52
Mn-54	0.02	0.02	0.02	0.02	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.03	0.04	0.02	0.02	0.04	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.22	0.04	Use RHR Pit Factor
Fe-55																									N/A	Use RHR Pit Factor
Co-60	0.20	0.05	0.02	0.02	0.06	0.04	0.07	0.04	0.14	0.07	0.04	0.04	0.04	0.07	0.02	0.04	0.03	0.04	0.14	0.08	0.03	0.03	0.03	0.03	0.06	Use Sample Avg.
Ni-63																									N/A	Use RHR Pit Factor
Sr-90	0.01	0.01	0.01	0.02	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.03	0.01	0.01	0.01	0.01	0.01	0.01	0.01	Use Sample Avg.
Nb-94	0.02	0.02	0.02	0.02	0.03	0.03	0.02	0.02	0.02	0.02	0.03	0.02	0.03	0.01	0.02	0.03	0.02	0.03	0.02	0.02	0.03	0.02	0.02	0.02	0.02	Use RHR Pit Factor
Tc-99	0.64	0.64					0.63	0.61					0.74	0.73					0.71	0.68					0.67	Use RHR Pit Factor
Ag-108m	0.02	0.02	0.02	0.01	0.02	0.03	0.02	0.02	0.02	0.02	0.03	0.02	0.02	0.02	0.02	0.03	0.02	0.03	0.02	0.02	0.02	0.02	0.02	0.02	0.02	Use RHR Pit Factor
Cs-134	0.03	0.02	0.02	0.02	0.04	0.04	0.03	0.03	0.03	0.03	0.04	0.03	0.04	0.02	0.02	0.04	0.03	0.04	0.03	0.03	0.04	0.03	0.03	0.03	0.03	Use RHR Pit Factor
Cs-137	0.20	0.04	0.02	0.02	0.03	0.03	0.07	0.05	0.08	0.03	0.03	0.03	0.04	0.04	0.03	0.04	0.03	0.23	0.06	0.06	0.03	0.03	0.03	0.02	0.05	Use Sample Avg.
Eu-152	0.05	0.05	0.05	0.04	0.07	0.08	0.07	0.07	0.06	0.07	0.08	0.07	0.07	0.06	0.06	0.08	0.07	0.09	0.07	0.06	0.07	0.07	0.06	0.06	0.07	Use RHR Pit Factor
Eu-154	0.06	0.05	0.06	0.05	0.10	0.08	0.08	0.07	0.08	0.07	0.12	0.10	0.10	0.06	0.06	0.11	0.08	0.10	0.08	0.08	0.09	0.08	0.09	0.08	0.08	Use RHR Pit Factor
Eu-155	0.06	0.05	0.06	0.05	0.08	0.10	0.08	0.07	0.08	0.07	0.08	0.08	0.06	0.07	0.06	0.09	0.07	0.09	0.08	0.07	0.07	0.08	0.08	0.07	0.07	Use RHR Pit Factor
Pu-238																									N/A	Use RHR Pit Factor
Pu-239																									N/A	Use RHR Pit Factor
Pu-241																									N/A	Use RHR Pit Factor
Am-241	0.07	0.10	0.08	0.07	0.11	0.21	0.14	0.10	0.15	0.10	0.05	0.17	0.04	0.12	0.07	0.07	0.10	0.14	0.10	0.11	0.12	0.15	0.11	0.15	0.11	Use RHR Pit Factor
Cm-243																									N/A	Use RHR Pit Factor

TABLE 3

Containment Floor & Wall Samples

	Sample # 175		Sample 176		Sample 177		Sample 178		Sample 179		Sample 180		Containment Sump Sample # 185								Containment Sump Sample 186								
Radionuclide	175-1C-01	175-1C-02	176-1C-01	176-1C-02	177-1C-01	177-1C-02	178-1C-01	178-1C-02	179-1C-01	179-1C-02	180-1C-01	180-1C-02	185-1C-01	185-1C-02	185-1C-03	185-1C-04	185-1C-05	185-1C-06	185-1C-07	185-1C-08	186-1C-01	186-1C-02	186-1C-03	186-1C-04	186-1C-05	186-1C-06	186-1C-07	Average Scaling Factor (to Co-60) for Containment Floor/Sump Surface Samples	Average of Surface Samples Diluted Over Total Depth
	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g		pCi/g
H-3													1400.00								1170.00							Use Samp Avg	1285.00
C-14	720.00	**0.52**	350.00	**0.50**									70.00	**0.50**	**0.51**						25.40	**0.57**	**0.54**					27.087	N/A
Mn-54	**0.10**	**0.06**	**0.07**	**0.02**	**0.07**	**0.04**	**0.06**	**0.03**	**0.09**	**0.10**	**0.02**	**0.02**	**0.38**	**0.03**	**0.03**	**0.03**	**0.04**	**0.03**	**0.03**	**0.02**	**0.13**	**0.02**	**0.05**	**0.03**	**0.07**	**0.03**	**0.02**	**0.010**	N/A
Fe-55													74.00								10.20							0.226	N/A
Co-60	7.78	**0.07**	23.10	**0.02**	6.43	**0.04**	3.98	**0.03**	5.20	0.15	1.68	0.03	240.00	0.38	0.09	0.08	0.11	**0.03**	**0.03**	0.10	70.90	**0.02**	**0.07**	**0.08**	**0.10**	**0.03**	**0.03**	Use Samp Avg	0.67
Ni-63													415.00								1620.00							12.289	N/A
Sr-90													20.10								0.95							0.049	N/A
Nb-94	**0.07**	**0.05**	**0.06**	**0.02**	**0.05**	**0.03**	**0.04**	**0.03**	**0.07**	**0.09**	**0.02**	**0.02**	**0.29**	**0.03**	**0.02**	**0.03**	**0.04**	**0.02**	**0.03**	**0.02**	**0.09**	**0.02**	**0.04**	**0.02**	**0.07**	**0.03**	**0.02**	**0.007**	N/A
Tc-99	**0.66**	**0.68**	**0.63**	**0.63**									2.84	**0.73**	**0.69**						**0.91**	**0.60**	**0.77**					0.034	N/A
Ag-108m	**0.11**	**0.04**	**0.07**	**0.02**	**0.09**	**0.03**	**0.08**	**0.02**	**0.09**	**0.07**	**0.03**	**0.01**	**0.57**	**0.04**	**0.02**	**0.03**	**0.03**	**0.02**	**0.02**	**0.04**	**0.03**	**0.02**	**0.03**	**0.02**	**0.06**	**0.02**	**0.02**	**0.011**	N/A
Cs-134	**0.11**	**0.06**	**0.08**	**0.02**	0.09	**0.04**	**0.06**	**0.04**	0.27	**0.12**	0.05	**0.02**	25.50	**0.04**	**0.03**	**0.04**	**0.05**	**0.03**	**0.03**	**0.03**	1.25	**0.02**	**0.06**	**0.04**	**0.08**	**0.03**	**0.03**	0.031	N/A
Cs-137	34.90	**0.05**	17.00	0.06	32.50	0.06	19.70	**0.03**	19.50	0.10	8.98	**0.02**	1270.00	6.02	0.15	0.01	0.12	0.03	0.12	3.52	584.00	1.59	**0.04**	0.10	0.16	0.03	**0.04**	Use Samples	2.69
Eu-152	**0.25**	**0.11**	**0.17**	**0.05**	**0.23**	**0.08**	**0.18**	**0.08**	**0.21**	**0.20**	**0.08**	**0.04**	**1.30**	0.15	**0.06**	**0.09**	**0.11**	**0.06**	**0.07**	**0.09**	**0.60**	**0.06**	**0.10**	**0.07**	**0.16**	**0.07**	**0.06**	0.028	N/A
Eu-154	**0.25**	**0.18**	**0.14**	**0.06**	**0.14**	**0.09**	**0.14**	**0.09**	**0.23**	**0.31**	**0.06**	**0.05**	**1.86**	0.08	**0.09**	**0.10**	**0.13**	**0.09**	**0.09**	**0.09**	**0.23**	**0.05**	**0.16**	**0.10**	**0.21**	**0.87**	**0.07**	0.024	N/A
Eu-155	**0.18**	**0.10**	**0.17**	**0.07**	**0.18**	**0.09**	**0.12**	**0.10**	**0.16**	**0.14**	**0.07**	**0.04**	**0.80**	**0.10**	**0.07**	**0.08**	**0.10**	**0.07**	**0.07**	**0.10**	**0.47**	**0.06**	**0.10**	**0.08**	**0.13**	**0.08**	**0.07**	0.021	N/A
Pu-238													5.08								0.63							0.015	N/A
Pu-239													1.92								0.24							0.006	N/A
Pu-241													54.80								9.86							0.184	N/A
Am-241	**0.27**	**0.15**	**0.27**	**0.07**	**0.24**	**0.13**	**0.07**	**0.20**	**0.24**	**0.07**	**0.13**	**0.07**	**7.06**	**0.14**	**0.09**	**0.05**	**0.08**	**0.09**	**0.10**	**0.22**	**0.74**	**0.08**	**0.20**	**0.17**	**0.20**	**0.11**	**0.16**	0.033	N/A
Cm-243													1.70								0.11							0.004	N/A

	Containment Floor				Containment Internal Walls						
Radionuclide	Duratek Sample 1/27/99 SML #1 First 0.5 inch	Duratek Sample 1/27/99 SML #1 0.5 to 1 inch	Duratek Sample 1/27/99 SML #1 1 to 1.5 inch	Avg Over All Samp at SML #1	Duratek Sample 1/27/99 SML #2 First 0.5 inch	Duratek Sample 1/27/99 SML #2 0.5 to 1 inch	Avg Over All Samp at SML #2	Duratek Sample 1/27/99 SML #3 First 0.5 inch	Duratek Sample 1/27/99 SML #3 0.5 to 1 inch	Duratek Sample 1/27/99 SML #3 1 to 1.5 inch	Avg Over All Samp at SML #3
	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g
Co-60	23.40	**1.00**	**0.58**	8.33	0.39	**0.50**	0.45	1.68	**0.23**	**0.52**	0.81
Cs-134	2.76	**0.68**	**0.64**	1.36	**0.63**	**1.40**	**1.02**	0.21	**0.41**	**0.66**	0.43
Cs-137	279.00	**0.49**	**0.76**	93.42	2.12	**1.10**	1.61	13.66	**0.60**	**0.58**	4.95

Notes: **1. Sample Results in Bold Type are <Minimum Detectable Activity (MDA)**
2. N/A - Not Applicable, Scaling Factors used to determine concentrations

Table 4

Residual Heat Exchanger (RHR) Pit Floor Samples

	Sample # 165						Sample # 166					Survey Area 2002			Survey Area 2008				
Radio-nuclide	165-1C-01	165-1C-02	165-3C-01	165-3C-04	165-5C-01	165-6C-01	166-1C-01	166-2C-01	166-3C-01	166-5C-01	166-7C-01	GTS Duratek Sample 1/27/99 SML #1 First 0.5 inch.	GTS Duratek Sample 1/27/99 SML #1 0.5 to 1 inch	GTS Duratek Sample 1/27/99 SML #1 1 to 1.5 inch.	GTS Duratek Sample 1/27/99 SML #1 First 0.5 inch.	GTS Duratek Sample 1/27/99 SML #1 0.5 to 1 inch	GTS Duratek Sample 1/27/99 SML #1 1 to 1.5 inch.	Average over Full Thickness	Scaling Factor to Co-60 for Outside Containment (Use Sample 165-1C-01 except as noted)
	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	
H-3	11.20	16.40	8.17	9.43	1.29	2.54	23.70	3.07	11.50	4.24	1.44							8.45	N/A use Average
C-14	0.77	0.45					0.76	0.77										0.07	0.011
Mn-54	0.18	0.03	0.04	0.02	0.03	0.04	0.07	0.03	0.02	0.03	0.03							0.01	0.003
Fe-55	49.90	3.31					3.87	4.10										2.34	0.737
Co-60	67.70	0.21	0.05	0.39	0.04	0.79	11.70	0.21	0.06	0.02	0.42	6.93	0.42	0.93	13.10	5.50	0.54	1.73	N/A use Average
Ni-63	21.80	2.09					23.70	1.43										1.98	0.322
Sr-90	4.59	0.03					1.74	0.01										0.28	0.068
Nb-94	0.13	0.03	0.04	0.02	0.03	0.03	0.06	0.03	0.02	0.02	0.02							0.01	0.002
Tc-99	0.86	0.50					0.79	0.82										0.07	0.013
Ag-108m	0.24	0.02	0.03	0.02	0.03	0.02	0.09	0.03	0.02	0.02	0.02							0.01	0.004
Cs-134	0.32	0.03	0.05	0.03	0.04	0.04	0.15	0.04	0.03	0.03	0.04				0.63	0.84	0.84	0.03	0.005
Cs-137	226.00	0.16	0.04	0.23	0.04	0.25	39.80	0.16	0.04	0.03	0.14	5.38	0.55	0.82	5.38	1.55	1.35	5.78	N/A use Average
Eu-152	0.59	0.07	0.10	0.06	0.07	0.07	0.23	0.09	0.06	0.06	0.07							0.04	0.009
Eu-154	0.29	0.09	0.12	0.06	0.09	0.08	0.90	0.09	0.08	0.09	0.09							0.05	0.004
Eu-155	0.45	0.06	0.10	0.06	0.07	0.06	0.19	0.11	0.07	0.07	0.07							0.03	0.007
Pu-238	0.76	0.07					0.92	0.07										0.07	0.011
Pu-239	0.21	0.05					0.28	0.05										0.02	0.003
Pu-241	11.90	4.11					7.94	2.96										0.86	0.176
Am-241	0.97	0.04	0.08	0.07	0.04	0.04	0.90	0.08	0.10	0.10	0.15							0.08	0.014
Cm-243	0.24	0.07					0.11	0.04										0.02	0.004

Table 5

RHR Pit Wall Samples

	Sample # 78							Sample # 171						Sample # 172						Sample # 173					Sample 174				
Radio-nuclide	78-C-1C-1	78-C-1C-2	78-C-1C-3	78-C-2C-1	78-C-3C-1	78-C-3C-2	78-C-3C-3	171-1C-01	171-1C-02	171-2C-04	171-3C-03	171-4C-03	171-5C-01	172-1C-01	172-1C-02	171-3C-02	172-4C-01	172-5C-01	172-5C-02	173-1C-01	173-1C-02	173-2C-03	173-3C-01	173-3C-02	174-1C-01	174-1C-02	174-4C-02	174-4C-01	Average All Wall Samples
	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g
H-3	12.3	25.9	25.8	3.27	20.6	18.4	5.62	13.80	17.40	1.50	1.61	7.92	2.64	12.10	8.77	1.04	1.54	1.56	1.55	6.50	6.42	1.48	1.31	1.43	7.71	N/C	6.20	N/C	7.66
C-14								0.72	0.64											0.74	0.67				0.64		0.74		N/A
Mn-54	0.092	0.0829	0.0829	0.0865	0.0762	0.0892	0.0931	0.02	0.04	0.03	0.03	0.03	0.03	0.02	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.02	0.04	0.02	0.06	0.03	N/A
Fe-55								2.72	3.36											3.78	4.94				4.02		4.50		N/A
Co-60	0.173	0.108	0.0936	0.125	0.0812	0.125	0.139	0.15	0.04	0.12	0.09	0.05	0.03	0.06	0.04	0.04	0.11	0.24	1.04	0.21	0.03	0.03	0.04	0.09	0.91	0.12	0.12	0.54	0.18
Ni-63								1.19	1.07											1.48	1.08				1.52		1.65		N/A
Sr-90	0.211	0.0696	0.0504	0.0575	0.0584	0.088	0.0904	0.02	0.02	0.01	0.02	0.01	0.01	0.01					0.03	0.09	0.02	0.01	0.01	0.01	0.67	N/C	0.04	N/C	0.07
Nb-94	0.081	0.076	0.0699	0.0782	0.0444	0.0786	0.0926	0.02	0.03	0.02	0.02	0.03	0.02	0.02	0.02	0.03	0.03	0.03	0.03	0.03	0.02	0.02	0.03	0.02	0.03	0.02	0.05	0.02	N/A
Tc-99								0.89	0.73											0.81	0.74				0.80		0.77		N/A
Ag-108m	0.013	0.0589	0.0691	0.0659	0.0579	0.0635	0.0506	0.02	0.03	0.02	0.02	0.02	0.02	0.02	0.02	0.02	0.02	0.03	0.03	0.02	0.22	0.02	0.03	0.02	0.03	0.02	0.05	0.03	N/A
Cs-134	0.279	0.0875	0.107	0.0993	0.0955	0.116	0.0128	0.03	0.04	0.03	0.03	0.03	0.03	0.03	0.04	0.04	0.04	0.03	0.04	0.03	0.03	0.03	0.03	0.03	0.05	0.03	0.07	0.04	N/A
Cs-137	7.59	0.0743	0.0905	0.0998	0.0746	0.111	0.104	0.23	0.06	0.24	0.14	0.05	0.05	0.02	0.03	0.03	0.12	1.03	1.06	0.53	0.04	0.02	0.04	0.55	1.03	0.05	0.10	4.08	0.63
Eu-152	0.303	0.178	0.225	0.221	0.174	0.233	0.194	0.06	0.08	0.06	0.07	0.08	0.06	0.06	0.08	0.07	0.07	0.10	0.07	0.06	0.06	0.06	0.08	0.07	0.09	0.06	0.15	0.09	N/A
Eu-154	0.256	0.194	0.207	0.288	0.213	0.255	0.261	0.07	0.09	0.08	0.08	0.10	0.08	0.07	0.10	0.10	0.07	0.08	0.08	0.09	0.08	0.08	0.09	0.07	0.09	0.07	0.19	0.07	N/A
Eu-155	0.272	0.18	0.181	0.219	0.168	0.214	0.224	0.07	0.07	0.07	0.07	0.10	0.07	0.07	0.08	0.08	0.76	0.11	0.08	0.06	0.08	0.07	0.10	0.08	0.08	0.06	0.15	0.09	N/A
Pu-238	0.014	0.0134				0.0234	0.0147	0.10	0.04											0.09	0.08				0.06		0.06		N/A
Pu-239	0.016	0.0133				0.0132	0.0259	0.02	0.03											0.08	0.06				0.01		0.03		N/A
Pu-241								2.32	2.74											2.64	2.72				2.74		2.48		N/A
Am-241	0.017	0.0411				0.0224	0.0352	0.04	0.04	0.13	0.10	0.20	0.10	0.12	0.13	0.13	0.11	0.16	0.12	0.02	0.06	0.10	0.14	0.14	0.05	0.09	0.05	0.12	N/A
Cm-243	0.031	0.0413				0.0225	0.0418	0.05	0.03											0.02	0.05				0.04		0.05		N/A

Table 6

Waste Disposal Building Basement Floors-Elevation 0'

Radionuclide	GTS Sample Dated 1/27/99 First 0.5 inch. pCi/g	GTS Sample Dated 1/27/99 0.5 to 1 inch. pCi/g	GTS Sample Dated 1/27/99 1 to 1.5 inch. pCi/g	Average over Total Thickness pCi/g
Co-60	160.55	**0.65**	**0.58**	2.79
Nb-94	0.28	**0.43**	<MDA	0.00
Cs-134	1.30	0.18	0.54	0.02
Cs-137	264.00	**0.72**	**0.52**	4.59
Eu-154	4.05	<MDA	<MDA	0.07
Eu-155	0.86	<MDA	<MDA	0.01
Am-241	11.03	<MDA	<MDA	0.19

Note: **1. Sample Results in Bold Type are <Minimum Detectable Activity (MDA)**

Table 7

Primary Auxiliary Building Sample (Other then RHR Pit)

	PAB Pipe Chase			
Radionuclide	GTS Sample Dated 1/27/99 First 0.5 inch.	GTS Sample Dated 1/27/99 0.5 to 1 inch.	GTS Sample Dated 1/27/99 1 to 1.5 inch.	Average over Total Thickness
	pCi/g	pCi/g	pCi/g	pCi/g
Co-60	34.10	**1.00**	**1.00**	1.42
Cs-134	5.18	0.13	**0.07**	0.22
Cs-137	74.00	**0.91**	0.28	3.08

Note: **1. Sample Results in Bold Type are <Minimum Detectable Activity (MDA)**

Table 8

Average Waste Concentration Calculation

Source of Concrete Waste	Estimated Waste Weight (Million lbs)	Contamination Levels Based On	Average Co-60 Concentration by Source (pCi/g)	Average Cs-137 Concentration by Source (pCi/g)	Average H-3 Concentration by Source (pCi/g)	C-14 Scaling Factor to Co-60	C-14 Concentration (pCi/g)	Mn-54 Scaling Factor to Co-60	Mn-54 Concentration (pCi/g)	Fe-55 Scaling Factor to Co-60	Fe-55 Concentration (pCi/g)	Ni-63 Scaling Factor to Co-60	Ni-63 Concentration (pCi/g)	Sr-90 Scaling Factor to Co-60	Sr-90 Concentration (pCi/g)	Nb-94 Scaling Factor to Co-60	Nb-94 Concentration (pCi/g)	Tc-99 Scaling Factor to Co-60	Tc-99 Concentration (pCi/g)	Ag-108m Scaling Factor to Co-60	Ag-108m Concentration
Containment Walls	40	Actual	0.06	0.05	5.48	2.522	0.143	0.003	0.000	0.737	0.042	0.322	0.018	Use Actual	0.011	0.0020	0.0001	0.0127	0.0007	0.0036	0.0002
Cont. Floor & Internal	20	Actual Floor	0.67	2.69	1285.00	27.087	18.105	0.010	0.006	0.226	0.151	12.289	8.214	0.0486	0.032	0.0072	0.0048	0.0343	0.0229	0.0112	0.0075
RHR Floors	1	Actual	1.73	5.78	8.45	0.011	0.020	0.003	0.005	0.737	1.272	0.322	0.556	0.0678	0.117	0.0020	0.0034	0.0127	0.0219	0.0036	0.0062
RHR Walls	2	Actual	0.18	0.63	7.66	0.011	0.002	0.003	0.000	0.737	0.130	0.322	0.057	Use Actual	0.073	0.0020	0.0003	0.0127	0.0022	0.0036	0.0006
Waste Disposal Walls	2.5	RHR Walls	0.18	0.63	7.66	0.011	0.002	0.003	0.000	0.737	0.130	0.322	0.057	RHR Walls	0.073	0.0020	0.0003	0.0127	0.0022	0.0036	0.0006
Waste Disposal Floors	0.5	Actual	2.79	4.59	8.45	0.011	0.032	0.003	0.007	0.737	2.058	0.322	0.899	0.0678	0.189	0.0020	0.0055	0.0127	0.0354	0.0036	0.0100
PAB Above El. 17.5'	7	Cont. Floor	0.67	2.69	7.66	0.011	0.008	0.003	0.002	0.737	0.493	0.322	0.215	0.0486	0.032	0.0020	0.0013	0.0127	0.0085	0.0036	0.0024
Fuel Pool Walls & Floor	1	RHR Floors	1.73	5.78	8.45	0.011	0.020	0.003	0.005	0.737	1.272	0.322	0.556	0.0678	0.117	0.0020	0.0034	0.0127	0.0219	0.0036	0.0062
Remainder of Fuel Bldg	8	RHR Walls	0.18	0.63	7.66	0.011	0.002	0.003	0.000	0.737	0.130	0.322	0.057	RHR Walls	0.073	0.0020	0.0003	0.0127	0.0022	0.0036	0.0006
Service Building	8	Cont. Walls	0.06	0.05	5.48	0.011	0.001	0.003	0.000	0.737	0.042	0.322	0.018	Cont. Walls	0.011	0.0020	0.0001	0.0127	0.0007	0.0036	0.0002
Misc Struct/Soil/Asphalt	10	Cont. Walls	0.06	0.05	5.48	0.011	0.001	0.003	0.000	0.737	0.042	0.322	0.018	Cont. Walls	0.011	0.0020	0.0001	0.0127	0.0007	0.0036	0.0002
Total	100	Weighted Avg. Conc.	0.284	0.974	261.88		3.68		1.67E-03		0.14		1.69		2.77E-02		1.25E-03		6.49E-03		2.04E-03

Source of Concrete Waste	Estimated Waste Weight (Million lbs)	Contamination Levels	Average Co-60 Concentration by Source (pCi/g)	Cs-134 Scaling Factor to Co-60	Cs-134 Concentration (pCi/g)	Eu-152 Scaling Factor to Co-60	Eu-152 Concentration (pCi/g)	Eu-154 Scaling Factor to Co-60	Eu-154 Concentration (pCi/g)	Eu-155 Scaling Factor to Co-60	Eu-155 Concentration (pCi/g)	Pu-238 Scaling Factor to Co-60	Pu-238 Concentration (pCi/g)	Pu-239 Scaling Factor to Co-60	Pu-239 Concentration (pCi/g)	Pu-241 Scaling Factor to Co-60	Pu-241 Concentration (pCi/g)	Am-241 Scaling Factor to Co-60	Am-241 Concentration (pCi/g)	Cm-243 Scaling Factor to Co-60	Cm-243 Concentration (pCi/g)
Containment Walls	40	Actual	0.06	0.0048	0.0003	0.0087	0.0005	0.0043	0.0002	0.0066	0.0004	0.0112	0.0006	0.0031	0.0002	0.1758	0.0099	0.0143	0.0008	0.0036	0.0002
Cont. Floor & Internal	20	Actual Floor	0.67	0.0312	0.0209	0.0277	0.0185	0.0236	0.0158	0.0214	0.0143	0.0150	0.0101	0.0057	0.0038	0.1837	0.1228	0.0332	0.0222	0.0043	0.0029
RHR Floors	1	Actual	1.73	0.0048	0.0082	0.0087	0.0150	0.0043	0.0075	0.0066	0.0114	0.0112	0.0193	0.0031	0.0053	0.1758	0.3034	0.0143	0.0246	0.0036	0.0062
RHR Walls	2	Actual	0.18	0.0048	0.0008	0.0087	0.0015	0.0043	0.0008	0.0066	0.0012	0.0112	0.0020	0.0031	0.0005	0.1758	0.0310	0.0143	0.0025	0.0036	0.0006
Waste Disposal Walls	2.5	RHR Walls	0.18	0.0048	0.0008	0.0087	0.0015	0.0043	0.0008	0.0066	0.0012	0.0112	0.0020	0.0031	0.0005	0.1758	0.0310	0.0143	0.0025	0.0036	0.0006
Waste Disposal Floors	0.5	Actual	2.79	0.0048	0.0133	0.0087	0.0243	0.0043	0.0121	0.0066	0.0185	0.0112	0.0313	0.0031	0.0086	0.1758	0.4908	0.0143	0.0398	0.0036	0.0100
PAB Above El. 17.5'	7	Cont. Floor	0.67	0.0048	0.0032	0.0087	0.0058	0.0043	0.0029	0.0066	0.0044	0.0112	0.0075	0.0031	0.0021	0.1758	0.1175	0.0143	0.0095	0.0036	0.0024
Fuel Pool Walls & Floor	1	RHR Floors	1.73	0.0048	0.0082	0.0087	0.0150	0.0043	0.0075	0.0066	0.0114	0.0112	0.0193	0.0031	0.0053	0.1758	0.3034	0.0143	0.0246	0.0036	0.0062
Remainder of Fuel Bldg	8	RHR Walls	0.18	0.0048	0.0008	0.0087	0.0015	0.0043	0.0008	0.0066	0.0012	0.0112	0.0020	0.0031	0.0005	0.1758	0.0310	0.0143	0.0025	0.0036	0.0006
Service Building	8	Cont. Walls	0.06	0.0048	0.0003	0.0087	0.0005	0.0043	0.0002	0.0066	0.0004	0.0112	0.0006	0.0031	0.0002	0.1758	0.0099	0.0143	0.0008	0.0036	0.0002
Misc Struct/Soil/Asphalt	10	Cont. Walls	0.06	0.0048	0.0003	0.0087	0.0005	0.0043	0.0002	0.0066	0.0004	0.0112	0.0006	0.0031	0.0002	0.1758	0.0099	0.0143	0.0008	0.0036	0.0002
Total	100	Weighted Avg. Conc.	0.28		4.89E-03		5.01E-03		3.81E-03		3.85E-03		3.69E-03		1.23E-03		5.09E-02		6.58E-03		1.11E-03

Table 9

Post Closure Dose Calculation

Radio-nuclide	Dose Equivalent per Concentration of Radionuclide - Resident Farmer (mrem/yr per pCi/g)	Weighted Average of All Waste (pCi/g)	Post Closure Dose for Avg of All Waste (mrem/yr)
H-3	4.202E-08	261.88	1.100E-05
C-14	0.000E+00	3.68	0.000E+00
Mn-54	0.000E+00	**1.67E-03**	0.000E+00
Fe-55	0.000E+00	0.14	0.000E+00
Co-60	0.000E+00	0.28	0.000E+00
Ni-63	0.000E+00	1.69	0.000E+00
Sr-90	0.000E+00	0.03	0.000E+00
Nb-94	0.000E+00	**1.25E-03**	0.000E+00
Tc-99	0.000E+00	6.49E-03	0.000E+00
Ag-108m	0.000E+00	**2.04E-03**	0.000E+00
Cs-134	0.000E+00	4.89E-03	0.000E+00
Cs-137	0.000E+00	0.97	0.000E+00
Eu-152	0.000E+00	5.01E-03	0.000E+00
Eu-154	0.000E+00	3.81E-03	0.000E+00
Eu-155	0.000E+00	3.85E-03	0.000E+00
Pu-238	4.299E-07	3.69E-03	1.587E-09
Pu-239	0.000E+00	1.23E-03	0.000E+00
Pu-241	0.000E+00	5.09E-02	0.000E+00
Am-241	0.000E+00	6.58E-03	0.000E+00
Cm-243	0.000E+00	1.11E-03	0.000E+00
		Total Post Closure Dose (mrem/yr)	**1.101E-05**

Note: **1. Values in Bold Type are based on Minimum Detectable Activity (MDA) (i.e. Radionuclide was not detected at the MDA concentration**

Docket No. 50-213
CY-05-002

Attachment 2

Haddam Neck Plant

Non-Radiation Worker Dose Calculations

January 2005

Dose to Non-Radiation Workers During Transport, Receipt, and Disposal

In order to assess the impact to non-radiation workers from the transport, receipt, processing, and disposal of low activity radioactive waste, an analysis was performed using the TSD-DOSE model (V 2.22)[2]. TSD-DOSE is a program developed by Argonne National Laboratory for estimating doses to facility workers and the surrounding public at Treatment, Storage, and Disposal (TSD) facilities from shipments of hazardous waste that may contain small amounts of radioisotopes.

The steps and parameters used to model the operations were chosen to be conservative yet realistic. In other words, engineering judgment and knowledge from several site visits was used to develop a model which could be applied to most TSD facilities and would produce conservative dose estimates in almost all cases (almost all because not every TSD facility was visited such that the conservatism of the model may not cover a site that has characteristics outside of the model). The default values were chosen to bound the TSD facilities visited.

TSD-DOSE estimates worker and public doses from seven operations. These operations can be turned on or off to reflect the actual TSD facility operations. In addition, many of the parameters used to model the typical operations can be adjusted to fit the actual facility. A dose is calculated for each operation based on radionuclide activities, waste characteristics, and any site-specific information entered by the user. Doses to various receptors are calculated by summing the doses from those operations that would potentially contribute to the exposure.

Ver 2.2 of the TSD-DOSE model was used to calculate the dose to the truck driver, the non-radiation worker at the TSD facility, and the public during transport and handling of the low activity material.

The worst-case scenario that will maximize the dose to the driver and the non-radiation worker at the TSD facility is a rolloff container with a 25 cubic yard capacity. In order to calculate the maximum dose to the non-radiation worker at the TSD facility for this worst-case scenario, the following assumptions for the seven operations in the TSD-DOSE model will be made. These assumptions are based on the actual Waste Control Specialists experience in handling similar waste streams.

Transport to the TSD facility.

The steps in this operation are: Load and secure shipment prior to transport; Drive loaded truck to TSD facility; Rest in back of cab en route to TSD facility; Maintenance (i.e. checking tires or refueling) of truck en route to TSD facility.

For trans-shipment from the generating facility to the rail transload facility, it is assumed that the driver is exposed for 2 hours at a distance of 10 feet from the waste. For bulk shipments the dose is insignificant to the railroad worker.

Receiving and sampling.

Weigh and survey truck and inspect manifest:

3931

1. "TSD-DOSE: A Radiological Dose Assessment Model for Treatment, Storage, and Disposal Facilities", Argonne National Laboratory, ANL/EAD/LD-4 (Rev. 1), September 1998.

One non-radiation workers for 30 minutes at the default distance of 5 feet.
Unload drums for inspection, sampling and storage prior to treatment:
This operation was not included
Inspect and sample drums.
This operation was not included
Transfer drums to storage awaiting treatment.
This operation was not included
Pump drummed liquids to storage.
Not applicable.

Storage.

Work in solid storage area.
This operation was not included
Transfer drums out of storage are for treatment.
This operation was not included
Work in liquid storage area.
Not applicable.

Incineration.

This operation was not included.

Treatment and on-site landfill.

Unload waste to mixing pit.
This operation was not included
Mix waste in mixing pit.
This operation was not included
Load truck and transport to landfill.
Two non-radiation worker for thirty minutes at a distance of 5 feet.
Unload truck at landfill.
Two non-radiation worker for 15 minutes at a distance of 5 feet.

Transport to off-site landfill.

This operation was not included.

Incinerator maintenance.

This operation was not included.

It is also likely that at least 10 different TSD facility workers could be exposed to any one shipment.

The results of the TSD dose calculation are attached.

Prepared by: Date: December 15, 2004

William P Dornsife

William P Dornsife, Corporate Radiation Safety Officer, Waste Control Specialists

TSD-DOSE: A Radiological Dose Assessment Model for Treatment, Storage, and Disposal Facilities

Version 2.22 - September 1998

Site: WCS
Shipment: Tranport and Disposal at WCS
User: dornsife

	TOTAL	EXTERNAL	INTERNAL
Dose to:			
Driver:	7.2E-05 mrem	7.2E-05 mrem	0.0E+00 mrem
Receiving worker:	9.1E-05 mrem	9.1E-05 mrem	0.0E+00 mrem
Incineration worker:	not applicable	not applicable	not applicable
Landfill worker:	1.5E-04 mrem	1.5E-04 mrem	0.0E+00 mrem
Offsite individual:	4.1E-09 mrem		
Offsite population:	1.4E-08 p-rem		
Worker Population:	4.6E-07 p-rem	4.6E-07 p-rem	0.0E+00 p-rem
Dose from:			
Transport to TSD facility:	7.2E-05 mrem	7.2E-05 mrem	not applicable
Receiving and sampling waste:	9.1E-05 mrem	9.1E-05 mrem	0.0E+00 mrem
Storage before processing:	0.0E+00 mrem	0.0E+00 mrem	not applicable
Incineration of waste:	not applicable	not applicable	not applicable
Burial at onsite landfill:	1.5E-04 mrem	1.5E-04 mrem	0.0E+00 mrem
Transport to offsite landfill:	not applicable	not applicable	not applicable
Incinerator maintenance:	not applicable	not applicable	not applicable

Doses due to each isotope (mrem - population dose in p-rem).

Isotope	Co60	Cs137+D	Fe55
Activity	5.4E-06 Ci	1.9E-05 Ci	2.7E-06 Ci
Release Fraction	1.00E-02	2.00E-03	5.00E-03
Driver	4.1 E-05	3.1 E-05	0.0 E+00
Receiving worker	5.2 E-05	3.9 E-05	0.0 E+00
Incineration worker	not applicable		
Landfill worker	8.5 E-05	6.3 E-05	0.0 E+00
Offsite individual	1.0 E-09	3.1 E-09	1.1 E-13
Offsite population	3.5 E-09	1.1 E-08	3.7 E-13
Worker population	2.6 E-07	2.0 E-07	0.0 E+00

Site Description

Operations included:

Transport to TSD facility
Receiving and sampling waste
Storage before processing
Burial at onsite landfill

Operations excluded:

Incineration of waste
Transport to offsite landfill
Incinerator maintenance

Parameters

The following are the adjustable parameters used to model each operation.
A (D) after a value indicates the default value was used.

Fraction solid waste = 1.000
Fraction liquid waste = 0.000
Pre-processed waste density = 1.0 E+00 g/cc
Post-processed waste density = 1.4 E+00 g/cc

Transport to TSD facility (4 steps)
Number of Workers: 1.0E+00 (D)
Truck bed dimensions (for all steps)
length: 2.00E+01 feet
width: 7.50E+00 feet
height: 4.50E+00 feet

Step A: Load and secure shipment
average distance: 3.00E+00 feet (D)
duration: 0.00E+00 hours
shielding thickness: 6.25E-02 inches (D)

Step B: Drive
average distance: 1.00E+01 feet
duration: 2.00E+00 hours
shielding thickness: 1.25E-01 inches (D)

Step C: Rest
average distance: 2.00E+00 feet (D)
duration: 0.00E+00 hours
shielding thickness: 1.25E-01 inches (D)

Step D: Maintenance in transit
average distance: 3.00E+00 feet (D)
duration: 0.00E+00 hours
shielding thickness: 6.25E-02 inches (D)

Receiving and sampling waste (5 steps)
Number of Workers: 1.0E+00

Step A: Weight truck, inspect manifest
average distance: 5.00E+00 feet (D)
duration: 5.00E-01 hours

Receiving and sampling waste (cont'd)

Step B: Unload drums
average distance: 3.00E+00 feet (D)
time per drum or pallet: 0.00E+00 hours

Step C: Inspect and sample drums
average distance: 5.00E-01 feet (D)
time per drum: 0.00E+00 hours
airborne respirable dust concentration: 1.0E+01 mg/m3 (D)
respiratory protection factor: 1.0E+01 (D)

Step D: Transfer solids to storage
average distance: 3.00E+00 feet (D)
time per drum or pallet: 0.00E+00 hours

Step E: Pump drummed oil to storage tank
average distance: 5.00E-01 feet (D)
time per drum: 0.00E+00 hours

Storage before processing (3 steps)

Step A: Workers in solid waste storage area
average distance: 3.00E+00 feet (D)
duration: 0.00E+00 hours

Step B: Transfer solids out
average distance: 3.00E+00 feet (D)
time per drum or pallet: 0.00E+00 hours

Step C: Workers in liquid waste storage area
average distance: 3.00E+00 feet (D)
duration: 0.00E+00 hours
shielding thickness: 1.25E-01 inches (D)
Storage tank dimensions:
length: 7.00E+00 feet (D)
width: 7.00E+00 feet (D)
height: 1.20E+01 feet (D)

Burial at onsite landfill (4 steps)

Number of Workers: 2.0E+00
Dump truck bed dimensions for steps A, C, and D):
length: 2.00E+01 feet
width: 7.50E+00 feet
height: 4.50E+00 feet

Step A: Unload waste to mixing pit
average distance: 5.00E+00 feet (D)
duration: 0.00E+00 hours
shielding thickness: 1.25E-01 inches (D)
airborne respirable dust concentration: 1.0E+00 mg/m3 (D)
respiratory protection factor: 1.0E+00 (D)

Step B: Mix waste in mixing pit
average distance: 1.00E+01 feet (D)
duration: 0.00E+00 hours
cover thickness: 2.00E+00 inches (D)
Mixing pit dimensions:
length: 1.00E+01 feet (D)
width: 1.00E+01 feet (D)
depth: 1.00E+01 feet (D)
cover thickness: 2.00E+00 inches (D)

Burial at onsite landfill (cont'd)

Step C: Load truck and transport to landfill
average distance: 5.00E+00 feet (D)
duration: 5.00E-01 hours
shielding thickness: 1.25E-01 inches (D)

Step D: Unload truck at landfill
average distance: 5.00E+00 feet (D)
duration: 2.50E-01 hours (D)
shielding thickness: 1.25E-01 inches (D)