CONNECTICUT YANKEE ATOMIC POWER COMPANY

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

Docket No. 50-213
CY-04-131

Re: 10 CFR 50.90

DEC - 1 2004

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Haddam Neck Plant
License Amendment Request, Use of a Basement Fill Model (Revising the Buried Debris Dose Model), and a Revision to Surface Contamination Release Limits for Various Piping Sizes

Connecticut Yankee Atomic Power Company (CYAPCO) has determined that the revised method of calculating the future ground water dose resulting from buried concrete from the current "Buried Concrete Debris Model" to a "Basement Fill Model" and a revision to surface contamination release limits for various piping sizes will require NRC review and approval. Therefore, pursuant to 10 CFR 50.59(c)(2), CYAPCO requests that the NRC review and approve the changes to the Haddam Neck Plant (HNP) License Termination Plan (LTP) through an amendment to Operating License, DPR-61, pursuant to 10 CFR 50.90. CYAPCO proposes to:

1. Modify the dose model for volumetrically contaminated concrete, rebar (hereafter referred to as simply "concrete"), the containment liner and embedded piping in basements that are to remain in place at the HNP site. The revised approach results in the offsite disposal of a larger percentage of the concrete structures (approximately 75% of that which would remain under the current approach). The overall effect results in a smaller amount of radioactivity contained in concrete to remain on-site than is allowed by the current LTP. CYAPCO intends to use the modified dose model which utilizes an inventory based approach to facilitate a revised remediation strategy for the containment building and other on-site buildings for which the basements are to remain on-site.

   The method of calculating the future groundwater pathway dose using the concrete debris model is being revised to an inventory based approach which

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will include activity inventories from the containment liner, embedded piping inside surfaces and radioactivity released from volumetrically contaminated concrete (which is controlled by diffusion rate through basement walls and flowable fill). The concrete that will remain is in the containment lower walls and floor mat, the in-core instrumentation sump, and the lower walls and floor of the spent fuel pool in the fuel building. The Basement Fill Model will also be used for other basements and footings that will remain on site using the results of future characterization surveys.

2. Additionally, CYAPCO proposes to include surface contamination release levels for other pipe diameters that may be encountered during the decommissioning beyond that currently included in the LTP for 4 inch piping.

Attachment 1 provides a discussion of the proposed changes, technical analysis, regulatory analysis (including No Significant Hazards Consideration Discussion), and environmental consideration. Attachment 2 provides estimates for release of radionuclides from potentially contaminated concrete at the HNP. Attachment 3 provides Kd values for backfill material for the HNP. Attachment 4 provides a marked-up version of the appropriate pages of the current HNP LTP.

The amendment request does not impact the public health and safety and does not involve a Significant Hazards Consideration (SHC) pursuant to the provisions of 10 CFR 50.92 (See SHC provided in Attachment 1).

The Independent Review and Audit Committee (i.e., Offsite Review Committee) has reviewed the amendment request and concurred with the determination.

The current decommissioning schedule calls for the release of the Containment for demolition in June 2005. Therefore, CYAPCO requests NRC approval of the license amendment by the end of May 2005.

In accordance with 10 CFR 50.91 (b), a copy of this license amendment request is being provided to the State of Connecticut.

There are no regulatory commitments contained in this submittal.

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

Sincerely

CONNECTICUT YANKEE ATOMIC POWER COMPANY

12/1/04

Wayne A. Norton
President

Date

Attachments:
1. Technical Analysis and Regulatory Analysis including Significant Hazards Consideration
2. Estimates for Release of Radionuclides from Potentially Contaminated Concrete at the Haddam Neck Plant
3. Kd Values of Backfill Material for Connecticut Yankee
4. Marked-up Pages of the HNP LTP

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

Subscribed and sworn to before me

This 1st day of December, 2004

Gerard P. van Noordennen

Date Commission Expires: 12-31-2007

Docket No. 50-213
CY-04-131

Attachment 1

Haddam Neck Plant
License Amendment Request, Use of a Basement Fill Model (Revising the Buried Debris Dose Model), and a Revision to Surface Contamination Release Limits for Various Piping Sizes
Technical Analysis and Regulatory Analysis Including Significant Hazards Consideration Discussion

December 2004

1.0 INTRODUCTION

Connecticut Yankee Atomic Power Company (CYAPCO) has determined that the revised method of calculating the future ground water dose resulting from buried concrete from the current "Buried Concrete Debris Model" to a "Basement Fill Model" and a revision to surface contamination release limits for various piping sizes will require NRC review and approval. Therefore, pursuant to 10 CFR 50.59(c)(2), CYAPCO requests that the NRC review and approve the changes to the Haddam Neck Plant (HNP) License Termination Plan (LTP) through an amendment to Operating License, DPR-61, pursuant to 10 CFR 50.90. CYAPCO proposes to:

1. Modify the dose model for volumetrically contaminated concrete, rebar (hereafter referred to as simply "concrete"), the containment liner and embedded piping to remain in basements that are to remain in place at the HNP site. The revised approach results in the offsite disposal of a larger percentage of the concrete structures (approximately 75% of that which would remain under the current approach). The overall effect results in a smaller amount of radioactivity contained in concrete to remain on-site than is allowed by the current LTP. CYAPCO intends to use the modified dose model which utilizes an inventory based approach to facilitate a revised remediation strategy for the containment building and other on-site buildings for which the basements are to remain on-site.

   The method of calculating the future groundwater pathway dose using the concrete debris model is being revised to an inventory based approach which will include activity inventories from the containment liner, embedded piping inside surfaces and radioactivity released from volumetrically contaminated concrete (which is controlled by diffusion rate through basement walls and flowable fill). The concrete that will remain is in the containment lower walls and floor mat, the in-core instrumentation sump, and the lower walls and floor of the spent fuel pool in the fuel building. The Basement Fill Model will also be used for other basements and footings that will remain on site using the results of future characterization surveys.

2. Additionally, CYAPCO proposes to include surface contamination release levels for other pipe diameters that may be encountered during the decommissioning beyond that currently included in the LTP for 4 inch piping.

CYAPCO proposes to revise the remediation strategy for remaining concrete as a result of lessons learned in the remediation and survey of buildings at other decommissioning sites and the characterization of the HNP containment concrete. The revised strategy involves a more aggressive removal of the contaminated concrete above site elevation 17' 6" and leaving some additional activated concrete that is behind the liner in the below grade elevation of the In Core Instrumentation (ICI) Sump area. CYAPCO now intends to remove all of the interior contaminated and activated concrete to the containment liner. This approach will eliminate the need for expending excessive resources on identifying and remediating

contamination which may have penetrated cracks and crevices in the concrete to the liner and will leave behind a surface condition more suitable for radioactivity detection. Conversely, the removal of the activated concrete behind the liner requires very high resource expenditures relative to the dose saved because of the location and configuration of the material and exposure to groundwater. Therefore, the revised strategy calls for leaving more of this material in place.

Using this approach, CYAPCO will remove a higher volume of contaminated concrete from the containment than initially proposed in the LTP and thus will remove more of the contaminated concrete source term. The only region of the plant with any significant activated concrete to remain on site, per this proposed amendment, would be a one to two foot width of activated concrete behind the liner in the ICI sump walls below the neutron shield tank. This portion of the containment (i.e., the ICI sump) is a right circular cylinder approximately twenty feet high, located between twenty and forty feet below grade. CYAPCO has examined the costs and logistical hazards associated with the removal of this concrete and believes that the safest approach, most consistent with ALARA principles, is to leave the liner and the activated concrete behind the liner intact, while removing all of the activated concrete inside and above the liner. Additionally, the ICI Sump will be filled with flowable fill, a pumpable grout, thereby rendering the sump inaccessible and reducing the quantity of radioactivity that will diffuse through to the groundwater inside of the containment basement.

The information submitted herein demonstrates that CYAPCO will continue to comply with the radiological criteria for unrestricted use specified in 10 CFR 20.1402 by meeting the established dose criteria of 25 mrem/yr or less from all pathways.

2.0 BACKGROUND

When the HNP LTP was approved by the NRC in November of 2002 (Reference 11.1), the general plan (and associated dose model) for the decontamination and final status survey of the HNP site can be summarized as follows:

- Structures that contained residual radioactivity would be decontaminated to the LTP required DCGLs and a Final Status Survey (FSS) conducted. After any independent verification surveys conducted by the NRC and resolution of any NRC inspection comments on the FSS, the building could be demolished and the concrete debris used to backfill any basement that remains.

- The dose model for an area filled with concrete debris included a component of the total dose that corresponded to the leaching of radionuclides from contaminated concrete into the groundwater surrounding the concrete debris. This portion of the dose model assumed that all of the radioactivity contained on the concrete would leach from the debris and reach equilibrium with the concrete and

groundwater instantaneously. The site structure that resulted in the lowest DCGL using this approach was the containment. In this case, the containment was assumed to be filled to 3 feet below grade with contaminated concrete.

CYAPCO has now changed the above decommissioning approach for the HNP site. The current plan for most structures at the HNP is to demolish and remove from site all concrete and structural materials that are above 4 foot below the plant grade level for most structures. This material will be removed from the site to an appropriate disposal facility depending on its radioactivity and hazardous material characteristics. For certain selected structures, such as the Primary Auxiliary Building and the Waste Disposal Building, all concrete and structural materials including that in the deep basement, will be removed from site and disposed as waste. For building basements and footings that remain, the radionuclide content will be assessed and the potential dose contribution after release of the area from the NRC license will be included with any other dose pathways in demonstrating that the area meets the License Termination Rule criteria of 25 mrem/yr plus ALARA.

Through this approach, a large quantity of concrete and structural debris that could contain residual levels of radioactive material would no longer be buried. Instead this material will be sent to an approved disposal facility should it be shown to contain detectable quantities of licensed radioactive material. This analysis shows, that even with the increase in allowable concentrations in a small area of the containment basement to remain, the total quantity of residual radioactivity allowed to remain on-site after removal of site areas from the license is lower under CYAPCOs' current decommissioning approach when compared to the approach approved in the LTP in November 2002.

The original plan for decommissioning of the HNP was to demolish structures to an elevation corresponding to three feet below grade. CYAPCO now intends to demolish structures, including the containment building structure to an elevation corresponding to four feet below grade. However, as a result of lessons learned during the remediation of structures at other decommissioning facilities and the characterization of the HNP containment concrete, CYAPCO intends to pursue a more aggressive remediation strategy of removing additional interior below grade concrete from inside the containment building. CYAPCO now plans to remove the containment building interior below grade concrete down to the containment steel liner. This results in less contaminated concrete surface area and less potential for the need to pursue remediation of contamination in cracks and crevices. Removing concrete down to the liner also leaves a smoother surface that enhances the detection capability of survey instruments during the performance of the FSS.

As part of this demolition strategy, CYAPCO will leave the containment liner in place for the portion that is below 4 feet below grade. The HNP LTP (Section 3.4.1.3) states that activated portions of the remaining foundations in excess of the volumetric DCGLs will be removed. This would have required penetration of the containment liner and removal of large volumes of concrete. As stated above, the activated

concrete that resides inside the containment building above the liner will be removed. CYAPCO is now proposing not to remove the remaining activated concrete inventory below the containment liner which makes up approximately 5% of the total contaminated containment interior concrete volume. As described in section 4 of this Attachment, this remaining activated concrete is located primarily in the walls and floor of the ICI Sump from the sump floor up to the former location of the bottom of the Neutron Shield Tank (NST), behind the steel liner (See Figure 3.2 in Section 3.1 of this Attachment). Removal of this activated concrete would be unnecessarily costly, resource intensive and hazardous.

The activated concrete activity concentrations from the ICI Sump are higher than the Concrete Debris DCGLs from the current LTP. If this activity concentration inventory were used in the current LTP, the future groundwater dose from the concrete would increase. Rather than assume that 100% of the activity in the concrete is released instantly, a conservative release rate has been calculated for this Basement Fill Model. When the release rate is related to groundwater concentration and dose, a larger activity concentration is allowed to remain in a limited area of the subsurface structures. However, using future groundwater dose values calculated in the analysis contained in this amendment request, the total activity to remain at the site is lower than that allowed in the current LTP. This is further described in later sections of this Attachment.

As noted in the Section 1.0 above, an additional change is described in this submittal which establishes additional allowable surface contamination levels for buried pipe to be released. These additional values allow for various piping sizes which may be encountered during decommissioning. Section 5.0 of this Attachment also includes the proposed changes to the groundwater dose modeling to address future groundwater dose. Section 7.0 of this Attachment presents the proposed buried piping release values for various piping sizes which may be encountered during the decommissioning.

The proposed decommissioning strategy has been evaluated for its effect upon the final state of the site and associated impacts on dose assessment, survey design and environmental assessment. CYAPCO has evaluated the dose significance of leaving a small area of the containment with higher concentrations than the Concrete Debris DCGLs contained in the current LTP and has concluded that the dose to the critical group, the resident farmer, is within the dose based NRC radiological criteria for license termination. Survey design considerations have been evaluated and are described in Section 6.0, Final Status Survey Considerations section of this Attachment.

3.0 REMEDIATION STRATEGY

3.1 Physical Description of Areas Containing Activated Concrete and Liner

a. The reactor pressure vessel (RPV) was enclosed and shielded by a combination of the primary shield wall and the ICI sump. The major arrangements of these structures are illustrated in Figure 3-1. The RPV is further shielded by the neutron shield tank (NST) which is supported inside the ICI sump.

b. The ICI sump area is a right-circular cylinder, approximately 20 feet tall. The diameter is approximately 25 feet for the top 5 feet from elevation -0 feet, 6 inches to elevation -5 feet 6 inches. The ICI sump area diameter narrows to 16 feet for the remaining 15 feet of depth from elevation -5 feet 6 inches to elevation – 20 feet 6 inches. The walls of the ICI sump vary in thickness but are a minimum of approximately 7 feet. The concrete mat below the steel liner of the ICI sump is 6 feet 6 inches thick.

c. The containment's carbon steel liner is attached to the inside walls of the ICI sump. The liner is covered by the NST (and grout) from an elevation of -5' 6" to an elevation of -0' 6'. The NST rested on a ledge in the ICI sump at elevation -5' 6". The walls (vertical sections) of the carbon steel liner are 3/8" thick; the floors (horizontal sections) are 1/4" thick. The ICI sump floor has approximately 1 foot of concrete on top of the liner, from elevation – 20 feet 6 inches to elevation – 19 feet 6 inches. On one side of the ICI sump, there is a horizontal tunnel (10' x 10') that provided access to the ICI sump area beneath the RPV. Figure 3-2 illustrates this arrangement, relative to the sump and RPV.

d. *The revised remediation plans include removal of all concrete in the* containment, inside of the containment liner above elevation -20 feet 6 inches. Thus, the remaining concrete with significant activity from activation would be located in the ICI sump walls, behind the liner and below the liner floor elevation that corresponds with the bottom of the Neutron Shield Tank. See Figure 3-2.

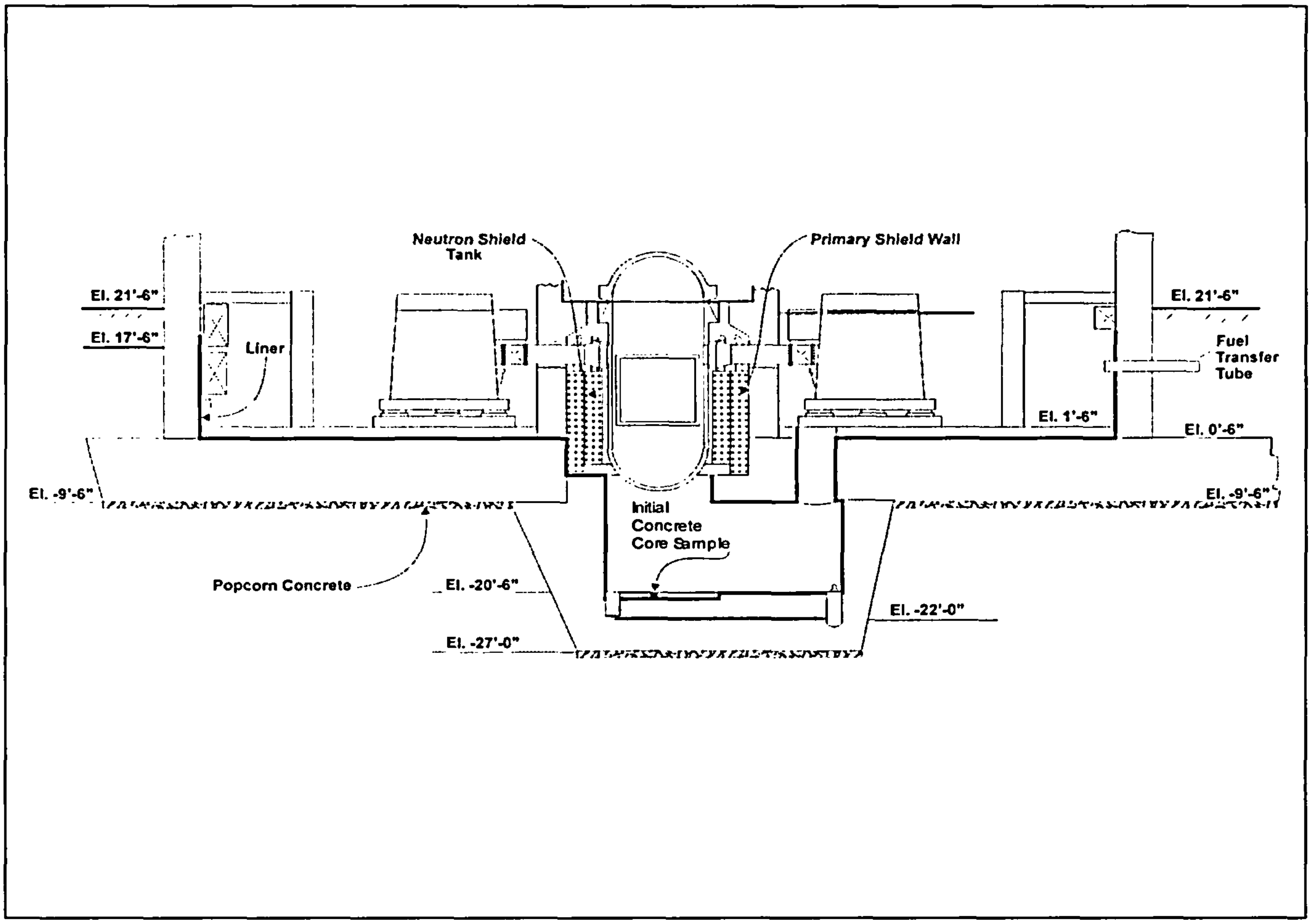

Figure 3-1 ICI Sump Area

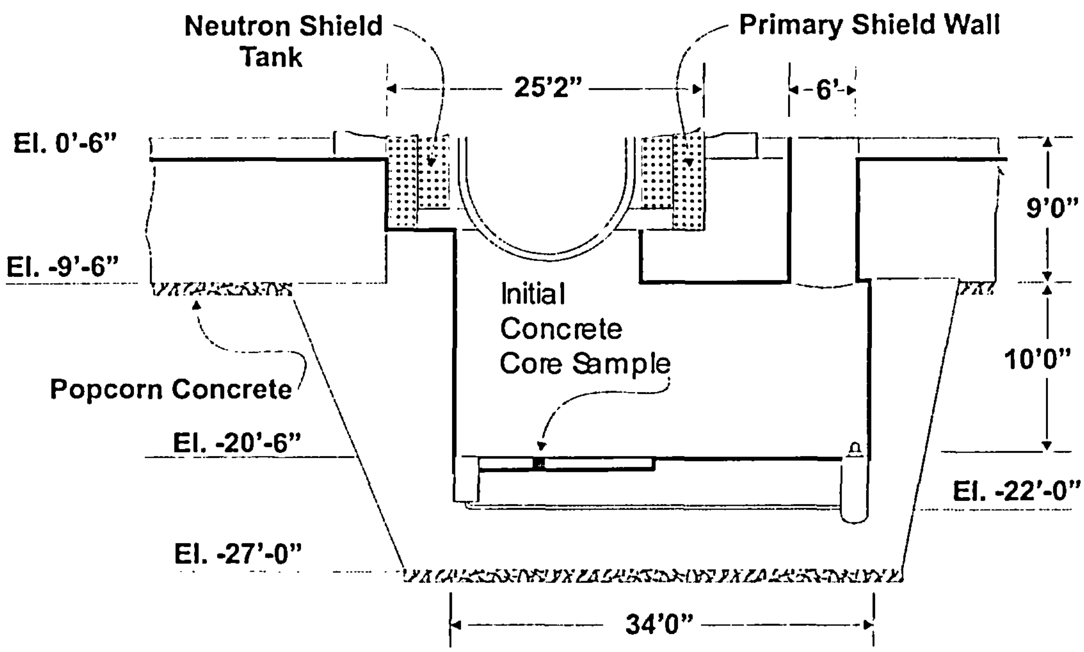

Figure 3-2 Detailed ICI Sump

3.2 LTP Activated Material Remediation Approach

The current LTP (Section 5.7.3.1.6) identifies activated concrete as a material for which some residual radioactivity would be left at the site. It was assumed that the activated concrete will be remediated to a value of the DCGLs developed throughout the relevant locations of the containment building. It was not anticipated that as much as two feet of concrete would need to be removed throughout the walls and floor of the ICI Sump as the characterization data to date appears to indicate (See Section 4.1 of this Attachment). CYAPCO had based its initial two feet expectation upon an activated concrete depth profile taken from a single concrete core bore to the containment liner at the expected highest flux location in the ICI sump floor (See Figure 3-2).

3.3 Revised Demolition Strategy

The revised demolition strategy is to remove the interior below grade contaminated and activated concrete down to the containment liner. Thus, all containment building interior concrete above the elevation -20' 6" would be removed. The activated concrete on the floor of the ICI Sump would also be removed. This approach would leave the activated concrete behind the liner of the walls and floor of the ICI Sump, which is a change from the current LTP plan of remediating the activated concrete to the DCGL values.

While the removal of all concrete above elevation -20' 6" results in a decrease in contaminated concrete inventory, leaving the activated concrete behind the liner results in an increase in the activated concrete activity inventory. As noted above and in Section 6.6.2 of the LTP, the final dose assessment will be based upon the inventory resulting from the actual post remediation level. Currently the DCGL for activated concrete is expressed as a concentration with units of pCi/g. In leaving the activated concrete behind the liner, the activated concrete activity concentration would consist of a distribution, and, based on the model developed for this amendment request, represents a total activity. From the characterization data it is clear that the concentration in a relatively small area of the containment basement and other relatively small areas of concrete, will be higher than the current LTP Buried Debris DCGLs.

3.4 Reasons for the Change to the Remediatation Strategy

CYAPCO has gained considerable experience with concrete demolition and the effectiveness of various remediation techniques

as decommissioning has progressed. Based on the experience and lessons-learned from earlier concrete demolition at other decommissioning facilities, a refined containment demolition plan was developed. Essentially, CYAPCO now plans to remove all buildings down to an elevation of 17' 6", 4 feet below grade elevation, and the containment building interior below grade concrete down to the containment liner, with the liner itself essentially intact. This results in less contaminated concrete surface area and less potential for the need to pursue remediation of contamination in cracks and crevices. Removing concrete down to the liner also leaves a smooth surface that enhances the detection capability of survey instruments. In addition, leaving the liner in place provides an additional barrier to the movement of radionuclides from the structural concrete into the basement fill volume although some holes will be made into the liner (total area of these holes is a small fraction of the total liner area) to allow groundwater to pass through the containment basement. As will be discussed later, no credit is taken for the liner in retarding the diffusion of radioactivity from the concrete into the containment groundwater.

3.5 Conclusions

CYAPCO has evaluated the difficulty, hazards and cost of removing the activated concrete behind the liner in the ICI Sump walls and has concluded that the safest approach, consistent with ALARA principles, is to leave the liner and the activated concrete behind the liner intact.

As shown in Section 5.5 of this Attachment, the revised approach (including leaving the concrete behind the liner in the In Core Sump in place) results in a total quantity of tritium (the highest activity radionuclide and that which affects future groundwater most) that is 80 % less than that which is allowed under the current LTP.

Additionally, the present plan is to fill the ICI sump volume with flowable grout to ensure the area remains inaccessible.

4.0 CHARACTERIZATION OF CONCRETE, REBAR, AND CONTAINMENT LINER

4.1 Summary of Currently Available Characterization Information

The characterization information obtained at the HNP indicates that there are potentially three mechanisms that introduce or create radioactivity in concrete structures that are to remain at the Haddam Neck Plant site. These mechanisms are:

- Surface or shallow contamination of metal or concrete surfaces due to plant operations such as leakage from plant systems
- Diffusion of tritium or soluble radionuclides deeper into the concrete due to the chemical makeup of the radionuclide. The sources of the radioactive material include water born pathways due to plant system leakage inside of structures and via groundwater passing by the outside of a concrete structure.
- Activation of material contained in the concrete matrix due to exposure to neutrons created by the fission process in the reactor vessel. This mechanism is localized and limited to the containment building at the HNP due to the very thick concrete walls that make up the outside of that building preventing any activation of other structures on site.

An additional mechanism involving an apparent diffusion of radionuclides in a gaseous form into concrete has been shown by characterization data to only apply to concrete inside of the containment liner at the HNP. As all of this concrete will be removed and disposed of as radioactive waste, this mechanism does not need to be considered in this license amendment request.

Concerning surface contamination, the methodology used to account for surface contamination will be discussed for the individual building areas in Section 5.0 of this attachment.

Concerning the volumetric contamination of concrete at the HNP, the characterization data has shown the following:

- Table 1 shows the results of characterization sampling for the containment mat below the liner. The results show the highest concentrations are generally directly below the liner and on the bottom of the mat. As discussed in Attachment 2, (the attached Brookhaven study), when the concentrations are higher on the surfaces of the concrete rather than uniform throughout the thickness, the average of the first 8 inches of sample should be used for the tritium determination and the results for the first wafer used for the other radionuclides detected. To provide the most conservative result the highest concentration determined from the following averages will be used in the dose calculation:

    - Average of all samples
    - Average of 1st inside and 1st outside samples
    - Average of first 2 inside and first 2 outside samples

It can also be seen from the sample data that the bedrock samples show no detectable radioactivity. Therefore, bedrock does not need to be included as a source in the groundwater dose calculation.

- The data in Table 2 indicates that the walls of the containment building outside of the containment liner are volumetrically contaminated with tritium and also show low levels of Co-60, Sr-90 and Cs-137. The analysis results for all the other radionuclides were less than detectable activity levels (detectable activity defined as activity greater than the 2 sigma error of the analysis). When the magnitude and consistency of the Sr-90 levels are considered, it is believed that these detections are false positives due to a bias in the laboratory analysis of the concrete cores. However until the Sr-90 values are determined by statistical analysis to be false positives, they will be considered as "actual" values in meeting the appropriate release criteria. The characterization data has shown the highest concrete concentrations to be on the side of the containment nearest the former location of the Refueling Water Storage Tank (RWST) and the outside tank farms. The historical information and characterization data suggest that leakage from the tanks in this area during plant operations caused soil contamination and subsequently groundwater contamination in this area. It appears that at least tritium was transported from the RWST area to the outside of the containment wall and then diffused into the containment concrete. For the radionuclides detected the average of the first wafers on the inside and the outside of containment results in the highest concentrations and will therefore be used in the dose calculation. The dose contribution from the cable vault portion of the containment has not been included in this analysis but will be added to the final calculation. Characterization data indicates this to be a relatively minor source.

- One core eight inches deep was obtained from the floor area beneath the former location of the reactor vessel in the ICI sump as shown on Figure 3-2. This sample was taken to assess activation of concrete in this area. This core was cut into 8 – 1" wafers. Table 3 displays the radionuclide analysis results for these samples. These results show elevated concentrations for tritium, Fe-55, Co-60, Eu-152 and Eu-154 throughout the depth of the core. C-14, Cs-134 and Cs-137 were also detected in low levels but are believed due to surface contamination and are not expected below the liner in significant concentrations. For conservatism, these radionuclides have been included in the dose calculation. As shown in Table 3, the concentration of

tritium increases as the depth into the concrete increases indicating that higher levels may be present under the containment liner in this area. Additional characterization sampling is planned in this area once the high dose rate commodities have been removed. This sampling will be from two elevations on the walls of the ICI sump to a depth of at least 3 feet and additional locations in the floor which will also be to at least 3 feet beyond the liner and will include rebar and liner samples. There is no concrete inside the liner on the walls of the ICI sump. The floor samples will be through the 8" of concrete above the liner, through the liner and at least 3 feet into the underlining concrete to complete the depth profile of the concrete in this location. Table 3, in addition to showing the results of ICI sump sampling, also shows a projection of the concentrations in the ICI walls which is calculated based on the ratios between floor and wall samples taken from the ICI sump at the Maine Yankee plant (Reference 2). This provides a reasonable approximation of the expected profile since the ICI Sump design for the two plants is very similar. The data in Table 3 also shows results for the other radionuclides. The contamination for these other radionuclides is on the surface or near surface and is not expected to be present in the concrete under the liner.

All the data discussed above is used to determine the expected groundwater concentrations resulting from buried concrete by multiplying the concentrations resulting from the unitized (1pCi/g) analysis shown in the Brookhaven diffusion study (Attachment 2) by the concentrations shown in Tables 1 through 3.

5.0 CALCULATION OF GROUNDWATER DOSE FROM BURIED CONCRETE, REBAR, PIPING AND THE CONTAINMENT LINER

5.1 Calculation Method – "Basement Fill Method"

The Basement Fill Model uses a total radioactivity inventory from buried plant structures and piping as its primary input. In this model the quantity of radioactivity released to the containment basement saturated zone (Volume below the water table) will be calculated individually for each source of radioactivity in the water table. The resulting groundwater concentration will be determined by calculating the equilibrium between the groundwater and the backfill soil using the results of a Brookhaven distribution coefficient (Kd) study of the HNP backfill as shown in Attachment 3. All of the individual groundwater dose values calculated from the different

sources will then be summed to determine the total future groundwater dose from the plant structures and piping buried for the containment building, the spent fuel building and footings to remain. There are many simplifying assumptions that will be discussed in the following sections. These simplifying assumptions result in a very conservative dose analysis for the future groundwater component. The method to be used to account for potential radioactivity on the portion of the containment liner and embedded piping to be left in place is discussed below. The footings of some of the buildings that are expected to remain on site will be assessed and any quantity of measurable activity calculated to be released using the Basement Fill Model will be included in the calculation of future groundwater dose for the containment basement. Any additional basements to remain after site release (i.e. "B" Switchgear and the Discharge Tunnels under the Turbine Building) will be assessed using the Basement Fill Model. For these additional basements, all of the released activity will be assumed to collect in the discharge tunnels for conservatism.

For other surface buildings that may remain, such as the EOF and Information Center, the Building Occupancy DCGLs from the current LTP will be applied. For these buildings, the concrete debris scenario or the basement fill model does not apply.

5.2 Groundwater Dose from Diffusion of Radioactivity in Concrete Basements

A study prepared by Brookhaven National Laboratory for the HNP (Attachment 2), establishes a methodology for the calculation of groundwater dose that results from radioactivity contained on the surface and volumetrically through concrete basement floors and walls as will be the case with subsurface concrete at the HNP. This calculation will include the inventory of measured radioactivity contained in any building footings that are to remain on site. This methodology is summarized as follows:

- The total inventory of radioactivity in concrete in the floors and walls (curies) is calculated based on an assumed concentration of 1 pCi/g for each of the radionuclides identified through extensive characterization.
- Next the Cumulative Fraction Released (CFR) for each radionuclide in each plant area is determined using worst case literature-based diffusion rates and the geometry of the concrete structure being analyzed.

- The CFR is next multiplied by the total inventory in the first bullet above to determine the total activity released to the soil/groundwater mixture inside of the containment basement.

- The resulting groundwater radionuclide concentrations are next determined by applying two adjustment factors:

1. A buildup factor that accounts for an increase in groundwater concentrations for radionuclides that diffuse relatively slowly compared to tritium.

2. A Retardation Factor that accounts for the adsorption of radionuclides on the backfill material in the basement. The Retardation Factor is calculated using the backfill soil distribution coefficients determined in Attachment 3. (Brookhaven study of Kds for backfill soil.)

   - The above calculation determines the unitized groundwater concentration (resulting from a 1 pCi/g concrete concentration) for each radionuclide that has been detected in concentrations higher than the 2 sigma error of the analysis.
   - As shown in Table 5, the groundwater concentrations corresponding to a 1pCi/g concrete concentration for each structure area are next compared to actual concrete concentrations using the characterization data displayed in Tables 1 through 3. Should detectable activity be assessed to be in any building footings to remain on site, the total inventory of that activity will be calculated and the yearly quantity released will be determined using the methodology defined in the Brookhaven Concrete Diffusion Study. This additional activity inventory will be included with all other sources, the containment basement (soil/groundwater mixture) and a future groundwater dose component calculated as for the other concrete sources above. These additional sources are expected to contribute little future groundwater dose although some footings may have concentrations that exceed the Concrete Debris DCGLs in small areas relative to the mass of concrete in the containment basement. Table 5 also includes a calculation of the groundwater dose from contamination expected to be present on the containment liner and embedded piping to remain after decontamination and final status survey is complete. The post decontamination contamination levels are based:

- o on an administrative Building Occupancy dose level of 10 mrem/yr from all radionuclides combined,
- o on radionuclide ratios based on the average of containment concrete sample data.

Due to the operational state of the Spent Fuel Pool, no characterization data has been obtained. The analyses in Table 5 uses values higher than are expected in the Spent Fuel Pit. This will be verified as discussed in Section 6.0 of this Attachment.

- It should be noted that although data has not yet been obtained for the radioactivity concentrations of activated metal (rebar and containment liner), experience at the Maine Yankee Plant (Ref.4) has shown the levels in activated rebar to be higher than the surrounding concrete. The additional characterization sampling, discussed earlier, will be conducted to confirm this assumption. Should the concentrations of certain radionuclides be higher in the concrete when compared to the metal, the higher concentrations will be used.

The available characterization data shows that the containment basement has the highest concentrations of the structures to be left on site. By evaluating this basement together with the spent fuel basement and buried footings, the highest possible groundwater dose due to buried structures will result. As discussed in the Brookhaven diffusion study, the method used is also very conservative due to the following simplifying assumptions:

- The diffusion rates used are the highest values from the range given in the literature
- By applying the Buildup Factor, the highest dose determined for each radionuclide is used even though these individual highest doses occur in different years.
- The radioactivity that diffuses from both the inside and the outside of the concrete in the containment mat and walls and the spent fuel pit floor and walls is assumed to migrate to the inside of the containment basement where it is diluted only by the volume of the initial fill of the containment with groundwater. This conservative assumption does not account for additional dilution that will occur from groundwater flow around and through the subsurface basements as penetration holes will be placed in these structures to encourage such flow. No credit is taken for the barrier to diffusion that the containment liner will provide. Although the liner will be pierced by holes that amount to a very small percentage of the liner area, it would still provide a substantial barrier to groundwater/concrete interaction for many years. As discussed below, the contamination left on the surface of the liner, on embedded

piping and in footings outside of containment is also included as a portion of the Basement Fill Model for the Containment basement.

- The Brookhaven Study assumes that 2.5 feet of grout will be placed above the activated concrete region of the In-Core Sump. The depth of the grout placed above the activated region will actually be 5 feet.

Groundwater concentrations calculated due to diffusion of radionuclides from concrete are shown in Table 5 for all of the radionuclides for which core sample results included detectable radioactivity over the 2 sigma error. The concentrations determined in Table 5 are summarized in Table 6-2 to show the expected future groundwater dose due to volumetric contamination for the containment and spent fuel pit. The calculated dose is 0.585 mrem/yr

Although the calculation shown is for the containment and spent fuel pit basements taken together, the same methodology will be used for other basements to remain at the site such as the Switchgear B building, the intake structure and the discharge tunnels/discharge structure under the turbine building as previously discussed. Routine Surveys and characterization sample results have shown these structures to have very low levels of residual radioactivity and are expected to result in lower groundwater concentrations than obtained in the containment/spent fuel pit calculation.

5.3 Groundwater Dose from Surface Contamination on the Containment Liner

As indicated previously, the containment liner will remain in place after the containment building is released from the license. This section describes how the groundwater dose from the surface contamination on the liner is calculated.

The process for demonstrating compliance for contaminated surfaces in the current LTP begins by determining the most restrictive DCGLs from the Building Occupancy Scenario or the Buried Concrete Scenario. As discussed earlier, the Buried Concrete Scenario is no longer applicable to the HNP and is being replaced with the Basement Fill Model. For the containment liner, the fill model assumes that contamination is present at levels corresponding to a total building occupancy dose based on an ALARA evaluation or administrative limit. For the purposes of this example calculation, a Building Occupancy dose of 10 mrem using an average radionuclide mix from the available concrete characterization data was used. The model further assumes the following:

- That the total inventory of radioactivity that corresponds to the contamination level that is present below the water table on the containment liner is released to the groundwater/backfill soil system present in the containment basement in the first year after release of this area from the NRC license.
- That the equilibrium groundwater concentration is calculated from the instantaneous release of the liner surface radioactivity and the radionuclide distribution coefficients (Kd) from the Brookhaven study of the HNP backfill soils.

This methodology is conservative as it assumes that all of the surface contamination is released in the first year to the groundwater and backfill soil. It is expected that some of the surface contamination would be retained on the liner surface or take more than a year to be released. Using this approach, the surveys conducted for this source will be performed using the MARSSIM guidance as described in Chapter 5 of the LTP for surfaces.

Groundwater concentrations calculated due to release of radionuclides from the containment liner surface are shown in Table 5 for all of the radionuclides for which core sample results included detectable radioactivity. The concentrations determined in Table 5 are summarized in Table 6-1 to show the expected future groundwater dose due to surface contamination on the containment liner. The calculated dose is 0.371 mrem/yr.

The above methodology applies to the containment liner. For all other buildings that have basement surfaces, no liner will remain in place. As discussed previously, the contamination on the surfaces of these basements are accounted for in the volumetric sampling and subsequently in the calculation of future groundwater dose due to volumetric contamination. Separate calculations of dose due to this surface contamination are therefore not required.

5.4 Groundwater Dose from Surface Contamination on the Embedded Pipes

The last source to be evaluated in determining the groundwater dose from buried structures and components is that resulting for the surface contamination contained on embedded piping that is to remain on site.

- Embedded piping is that which penetrates the containment or spent fuel pool basement and will not be removed or grouted. This source will be included in the calculation of the containment interior groundwater concentration as follows:

- As with the containment liner, embedded piping will be assumed to be surveyed to an Operational Building Occupancy DCGL that corresponds to the 10 mrem/yr administrative level using an average radionuclide mix from the available characterization data.
- The surface contamination levels determined in the last bullet (levels determined in Table 4 for the containment liner used for the purposes of this estimate) will be next multiplied by the embedded piping surface area to determine the total inventory for each radionuclide. The appropriateness of these radionuclide ratios will be confirmed by smear data that will be collected from the embedded piping to remain.
- This inventory is next assumed to be released to the containment basement groundwater/soil mixture within the first year after license termination. Using the backfill soil Kds, the groundwater concentrations for the various radionuclides are calculated as for the other sources above. The calculation of groundwater concentration from embedded piping is shown in Table 5 for each radionuclide.

As for the containment liner, this approach is conservative as it does not take credit for any contamination that is retained on the surface or that which takes more than one year to be released.

Groundwater concentrations calculated due to release of radionuclides from embedded piping surfaces are shown in Table 5 for all of the radionuclides for which core sample results included detectable radioactivity. Applying the Groundwater DCGLs, the doses determined in Table 5 are summarized in Table 6-1 to show the expected future groundwater dose due to surface contamination on the embedded piping. The calculated dose is 0.0218 mrem/yr.

5.5 Calculation of Total Groundwater Dose from all Concrete / Structures / Embedded Piping Sources

Sections 5.2 to 5.4 of this Attachment determine the future groundwater doses due to the individual sources, concrete, containment liner and embedded piping. The individual doses are next summed to determine the total future groundwater dose from all sources in Table 6-1. The total future groundwater dose from this calculation is 0.978 mrem/yr. As discussed earlier this calculated dose will be used to supply the "future groundwater" dose component of the compliance equation (5-7) included in Chapter 5 of the LTP. The currently estimated dose for the future

groundwater dose is not a significant fraction of the license termination criteria of 25 mrem/yr from all pathways.

As previously discussed this calculation uses estimates of concrete concentrations in areas that have not yet been characterized. There may also be certain footings that are assessed to contain residual radioactivity. When this characterization data has been obtained (See Section 6.0 of this Attachment) the future groundwater dose will be recalculated as part of the final status survey of the containment liner.

5.6 Comparison of Radioactivity to Remain to that Allowed by the Current LTP

A comparison has been performed between the quantity of contamination allowed under the current LTP (prior to the required ALARA evaluation) and the actual values that have been determined in the analysis contained in this submittal.

The allowable quantity of radioactivity under the current LTP has been calculated as follows:

- The weight of concrete that would have been left on site under the "Concrete Debris" scenario for the RCA buildings has been estimated at 120 million pounds in a recent internal estimate.
- The spent fuel building and the containment building make up the majority of the concrete that would have remained as debris under the Concrete Debris scenario. The "future groundwater" doses determined in Table 6-1 for the detected radionuclides in concrete are used in Table 6-2 to determine the fractions of the total future groundwater dose for those radionuclides. For some of the radionuclides, the fractions calculated by this method were very small (ranged from 4.4 E-05 for Am-241 down to 2.7 E-09 for Eu-154). These fractions were adjusted to values between 1E-03 and 5E-04 to show a more practical limit for these radionuclides while maintaining the resulting dose as an insignificant value (less than 0.1% of total dose)
- The LTP, Section 5.4.7.1 example for applying LTP Equation 5-6 was used as the dose component for "future groundwater" (Fractional Value = 0.5). This corresponds to a future groundwater dose of 12.5 mrem/yr. Using the fractions determined in the last bullet, allowable dose due to each radionuclide was determined. It can be seen from Table 6-1 that H-3 amounts to essentially the entire dose (99%) due to concrete contamination under the building debris scenario.

- Using the allowable dose of 12.5 mrem/yr, the Concrete Debris DCGLs and the fraction of total Concrete Debris dose that is due to water dependent pathways, the allowable concentration in the concrete for the Concrete Debris Scenario is determined and shown in Table 6-2.
- Finally the allowable concrete concentrations determined are multiplied by the total weight of concrete (120 million lbs) to determine the total allowable activity that could remain under the current LTP.

For the estimated activity to remain under the revised decommissioning approach, the inventory of activity in concrete is calculated as follows:

- As can be seen in the lower half of Table 6-2, the concrete concentrations for all detected radionuclides from Tables 1 through 3 are input to the table.
- These concentrations are next multiplied by the weight of concrete for each area to determine the total activity contained for each radionuclide in each area.
- Lastly all the area activities are summed to determine the total activity to remain under the revised decommissioning approach

Table 6-2 also contains the results of a calculation of the total activity in concrete to remain under the revised decommissioning strategy as a percent of that allowed under the current LTP. It can be seen from Table 6-2, that the activity of H-3 (which was equivalent to more than 99% of the groundwater dose due to concrete) expected to remain under the revised decommissioning approach, will be 80% less than that which would be allowed under the current LTP (prior to the required ALARA evaluation). Although other radionuclides such as Co-60 and Am-241 are projected to have activities higher than that allowable in the current LTP, the future groundwater dose resulting from these other radionuclides are approximately 0.005 mrem/yr and therefore insignificant. The conclusion is that considerably less total radioactivity will remain on-site under this revised approach than is allowed under the current LTP approach.

6.0 FINAL STATUS SURVEY CONSIDERATIONS

The changes in the method of calculating the future groundwater dose component of the LTP compliance equation necessitates changes to the surveys required as part of the Final Status Survey (FSS). The following describes the sampling and surveys that will be performed to

collect the data needed to perform the Basement Fill Model calculation of future groundwater dose.

6.1 Contaminated Surfaces

The final status survey requirements for metal surfaces such as the containment liner and embedded piping are contained in the current LTP and are unchanged in this revised approach except that only the Building Occupancy DCGLs will be used.

6.2 Contaminated Concrete

As previously discussed, the Basement Fill Model treats contaminated concrete as a volumetric source of radioactivity. It is therefore appropriate to utilize volumetric concrete sample results to determine the data to be used in the calculation. Table 7 (See below) shows the number of samples that have been taken to date and the minimum number of additional samples that will be taken to provide enough characterization data to allow the confident calculation of the future groundwater dose. The samples taken will be analyzed so that the profile with the depth of the concrete can be confidently shown. Except for the in core sump, the sampling will include analysis of concrete from the inside and outside surfaces and for areas inside the wall with at least 15% of the wall/floor thickness characterized.

The mechanism that has caused volumetric contamination in concrete is in many cases specific to certain radionuclides. Radionuclides such as H-3 and Sr-90 have been detected in concrete in contact with contaminated groundwater. Areas that have been subject to substantial neutron flux typically display H-3, Fe-55, Co-60, and Eu-152 among others. To adequately assess the volumetric contamination of concrete, a wafer from at least 20% of the locations listed in Table 5-10 of the HNP LTP will be analyzed for all 20 radionuclides listed in Table 2-12 of the HNP LTP. For radionuclides expected in certain areas of concrete, a sufficient number of wafers from all locations will be analyzed for the expected radionuclides to allow determination of a profile in the concrete at that location.

Table 7

Volumetric Concrete Sample Requirements

Basement Area to Remain (Below Elevation 17'6")	Concrete Samples Collected to Date (10/30/2004)	Additional Concrete Samples to be Collected	Minimum Total Number Of Samples To be Used for the Inventory Calculation
Containment Mat	8	6	14
Containment Walls	4	6	10
In Core Sump	1	8	9
Spent Fuel Pool	0	12	12
Cable Vault	7	6	13
"B" Switchgear Building	0	8	8
Discharge Tunnels/Structure	0	10	10
Intake Structure	2	6	8

As previously indicated, certain building footings that may remain on site will be sampled using concrete sampling (at least 3 samples per footing or group of footings) or assessed using the results of nearby soil sampling to determine their radioactivity content. If the assessment is done using soil samples, the following methodology will be used:

- The average of the soil samples result (for plant-related radionuclides detected) will be calculated for each footing (at least 3 samples per footing or group of footings).

- For Tritium:

    - Using the soil distribution coefficient data (Kds) shown in Table F-1 and equation 6-1 of the current LTP, the groundwater concentration in equilibrium with the soil concentrations in the last bullet will be determined
    - Using the concrete distribution coefficients shown in Table F-4 and equation 6-1 of the current LTP the concrete concentrations in equilibrium with the groundwater concentrations calculated in the last bullet will be determined.

- For radionuclides other than H-3:

    - At least six pairs of concrete samples with adjacent soil samples will be collected at locations affected by plant leakage and/or groundwater contamination
    - The average and %CV (Coefficient of Variation) of the ratios of concrete concentrations to soil concentrations will be calculated
    - If the %CV of the data is less than 25 %, the average ratio will be used to determine concrete concentrations from adjacent soil samples
    - If the %CV is 25% or greater more samples will be taken until a satisfactory variance is calculated or the worst case ratio will be used

The results of this sampling or assessment will then be used as input to the calculation of future groundwater dose using the basement fill model.

7.0 BURIED PIPING

Generally there are two types of piping that may remain at the HNP site, buried or embedded. Each of these types will be considered separately for FSS because of their resident locations. Typically, buried pipe will be located in the saturated subsurface areas and will be grouted and capped while embedded pipe will be found in penetrations which are parts of a structure and will not be grouted or capped. The method of showing compliance with the release criteria for embedded piping has been addressed in Section 5.4 of this Attachment. The following describes the release limits for buried piping.

7.1 Buried Piping

Buried piping is piping that is located in the saturated subsurface areas of the site. Contaminated buried piping following any required remediation and surveying will be grouted as agreed to with the State of Connecticut Department of Public Utilities Commission (DPUC).

An example of buried piping and the final disposition is the containment 18 inch diameter stainless steel Residual Heat Removal (RHR) Spray Recycle Suction line. This line runs from the containment sump downward approximately 10 feet through the containment concrete mat, makes a 90 degree turn and runs approximately 25 feet, makes a 45 degree turn and runs another 13 feet where it exits the north wall under the containment concrete mat where it runs 100 feet encased in concrete then passes through the east wall of the Primary Auxiliary Building. This line has been remediated using high pressure water lance techniques.

Following the demolition of the Primary Auxiliary Building, the piping will be exposed at the north containment wall; here it will be internally radiologically assessed to the containment mat sump pipe opening inside of the containment, a total of approximately 50 feet. Following the successful radiological assessment, the pipe will be grouted with concrete and capped.

7.2 Current Requirements of the License Termination Plan

The current LTP, Section 5.4.7.5, Release Limits for Non-Structural Components and Systems describes the release limits for buried piping.

7.3 Proposed Buried Piping Contamination Levels and Conditions

The proposed change will revise Section 5.4.7.5 of the LTP for the radionuclides listed in Table 5-7 of the LTP, Release Limits For Buried Piping.

The new Table (See Attachment 4) provides a listing of the radionuclides of concern and the limits based on the inside diameter of the pipe. The surface radionuclide contamination levels for the various piping sizes were determined using the piping surface area ratio to the 4 inch diameter pipe size. Scaling the surface contamination of the various piping sizes to the 4 inch diameter pipe size of the current LTP, the new Table continues to use the basis from the Health Physics Department Technical Support Document, BCY-HP-0105 Revision #: 1, Dose Evaluation of Buried Piping.

The total length of contaminated buried piping that will remain is a very small amount in length of piping and should not exceed a few hundred feet.

For buried piping in contact with the saturated zone, an analysis has been performed to determine surface activity limits for the remaining piping that will result in no more than a 1 mrem/yr dose as referenced above. This piping will be grouted with concrete (after any required remediation and surveying), as agreed to with the State of Connecticut. To simplify the analysis, the piping material is assumed to be eroded away, leaving the slug of grout that has transferred to it, the contamination from the interior surface of the piping. Consistent with these simplified assumptions, the DCGLs calculated in Chapter 6 of the LTP for concrete debris are used in developing the surface contamination limits for this piping.

In order to calculate the release limits for the piping (corresponding to 1 mrem/yr), first, for each radionuclide, the DCGL representing 25 mrem/yr from all pathways for concrete debris and the fraction of dose from the water dependent pathways were used to determine the volumetric limits from water dependent pathways only (as the buried piping is well below the soil surface, thus eliminating an external dose contribution, and is in contact with the groundwater). These limits are then normalized to represent a volumetric limit that would result in 1 mrem/yr. Finally, the volumetric contamination is converted to surface contamination for the various pipe diameters (because the larger the diameter, and subsequently the radius, the larger the surface activity limits can be). The release limits to be applied to this piping are given in the new Table (Attachment 4).

7.4 Justification for Adjustments to Buried Piping Contamination Levels

The radionuclides of concern have not been changed from the current LTP, Table 5-7.

The length of buried piping that will remain is a small amount in length and should not exceed more than a few hundred feet.

The total amount of surface contamination in the piping will be included in the release record for the associated survey unit to demonstrate that the area meets the release limits that correspond to a dose of 1 mrem/yr. The sum of the fractions is used for each of the radionuclides present.

The piping in the saturated zone is grouted following any needed remediation and therefore is not considered to be an external exposure or a source of contamination to the containment basement.

With the commitment to grout the buried piping and to assess the impact of the contamination on the groundwater the piping material is assumed to be eroded away leaving the slug of grout that has the surface contamination from the piping transferred to its surface. Because of the conservative simplifications and assumptions made, the DCGLs already calculated for concrete debris (which assume contact with the ground water) are used in developing the surface contamination limits.

7.5 Conclusion

The surface contamination levels for the various piping sizes when converted to the volumetric contamination based on the grouting of the piping does not change the concentrations being left. This is a result of using the volumetric limit that results in 1 mrem/yr dose in pCi/gm to scale the surface contamination limits for various piping sizes.

8.0 OTHER CONSIDERATION

8.1 ALARA

The proposed amendment request includes a future groundwater dose calculation (using the Basement Fill Model) based on characterization results obtained to date along with conservative estimates of concrete radionuclide concentrations for areas not yet sampled and contamination levels expected on the containment liner surface after final status survey. The results of this calculation are that the future groundwater dose is expected to be less than 1 mrem/yr which is a small fraction of the total license termination allowable dose of 25 mrem/yr. Additional characterization samples will be taken to confirm the current results and address areas not yet sampled so that the final future groundwater dose component of the LTP compliance equation can be calculated.

Based on this low future groundwater dose and high expense and hazard with performing additional remediation beyond that discussed in this Attachment, it fulfills the ALARA requirements of LTP Section 4.2 to leave small areas of activated or otherwise contaminated concrete, metal or rebar that exceed the concrete debris DCGL for the current HNP LTP. Therefore, no further ALARA evaluation is necessary.

9.0 REGULATORY SAFETY ANALYSIS

9.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.92, CYAPCO has reviewed the amendment request and concluded that the amendment request does not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The amendment request does not involve an SHC because the amendment request would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The activities included in the amendment request are within the bounds of those contained in the HNP Updated Final Safety Analysis Report (UFSAR). The HNP UFSAR Chapter 15 provides a discussion of the radiological events postulated to occur as a result of decommissioning activities with bounding consequences resulting from a resin container accident. This accident is expected to contain more potential airborne activity that can be released from other decommissioning events. The radionuclide distribution assumed for the resin container has a greater inventory of transuranics radionuclides (major dose contributor) than the distribution of plant derived radionuclides in the components involved in other decommissioning activities.

The HNP UFSAR also discusses a fuel handling accident in the fuel building, involving the drop of a spent fuel assembly on to the fuel racks. The postulated drop assumes the rupture of all fuel rods in the associated assembly. The probability or consequences of this accident would not be increased during any future fuel operations in the spent fuel building related to decommissioning. Transfer of the spent fuel to canisters for dry cask storage involves additional restrictions contained in the cask certificate of compliance in order to maintain decommissioning activities within the assumptions of and consequences of the fuel handling accident.

No systems, structures, or components that could initiate or to be required to mitigate consequences of an accident are affected by the amendment request in any way not previously evaluated in the HNP UFSAR. Therefore, the amendment request does not involve any increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   Accident analyses related to decommissioning activities are addressed in the HNP UFSAR. The activities included in the amendment request are within the bounds of those considered in the HNP UFSAR. Thus, the amendment request does not affect plant systems, structures, or components in any way previously evaluated in the HNP UFSAR. The amendment request does not introduce any new failure modes. Therefore, the amendment request will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The HNP LTP is a plan for demonstrating compliance with radiological criteria for license termination as provided in 10 CFR 20.1402. The margin of safety defined in the statement of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 merem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use (one of the criteria of 10 CFR 20.1402). This margin of safety accounts for the potential effects of multiple sources of radiation exposure to the critical group. Since the HNP LTP was designed to comply with the radiological criteria for license termination for unrestricted use, this license amendment request supports this margin of safety.

   Also, as previously discussed, the bounding accident for decommissioning is the resin container accident. Since the bounding decommissioning accident results in more airborne radioactivity than can be released from the other decommissioning events, the margin of safety associated with consequences of decommissioning accidents is not reduced by this amendment request.

   Thus, the amendment request does not involve a significant reduction in the margin of safety.

9.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.82(a)(11) establishes the criteria to be used by the NRC for terminating the license for a power reactor facility. These criteria include (1) dismantlement has been performed in accordance with the approved license termination plan and, (2) the final radiation survey and associated documentation demonstrate that the facility and site have met the criteria for decommissioning in 10 CFR 20, Subpart E. 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use," allows termination/amendment of license and release of a site for unrestricted use if the residual activity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of a critical group that does not exceed 25 merm/yr and the residual radioactivity has been reduced to levels that are ALARA. The technical analysis and conclusions of this submittal describe the methods used for conducting a dose assessment to develop the DCGLs for demonstrating compliance with the unrestricted use criteria in 10

CFR 20.1402. The proposed revised remediation strategy for the activated concrete and soil (surface and deep), demonstrates that the radiological criteria for license termination are met.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

10. ENVIRONMENTAL CONSIDERATION

CYAPCO has evaluated the amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance 10 CFR 51.22. CYAPCO has determined that the amendment request meets the criteria for categorical exclusion set forth in 10 CFR 51.22(C)(9) and as such, determined that no irreversible consequences exists in accordance with 10 CFR 50.92(b). This determination is based on the fact that the amendment request meets the following specific criteria.

(i) The amendment request involves no significant hazards consideration.

As demonstrated in Section 9.1 of this attachment, the amendment request does not involve an SHC.

(ii) There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

The environmental impacts associated with doses to members of the public as a result of decommissioning activities and site release for unrestricted use were considered in the Generic Environmental Impact Statement on Decommissioning Activities of Nuclear Facilities (NUREG-0586 and Supplement 1 to NUREG-0586) and the Generic Environmental Impact Statement in Support of the Rulemaking on Radiological Criteria for License Termination (NUREG-1496). In support of the HNP Post Shutdown Decommissioning Activities Report (PSDAR), Revision 2, (Reference 11.3) CYAPCO performed an environmental review of site specific decommissioning activities. It was concluded that the environmental impacts associated with the site specific decommissioning activities will be bounded by appropriate previously issued environmental impact statements. In particular,

the decommissioning activities covered by the HNP LTP will result in radiation doses to the public below a comparable level when the plant was operating. Radiation dose to the public will be minimal. The release of effluents will continue to be controlled by plant procedures throughout decommissioning. CYAPCO will continue to operate in accordance with the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) during decommissioning activities. In addition, because of the decay of short-lived radionuclides, the types of nuclides that could potentially be released in effluents have decreased.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

In support of the HNP PSDAR, Revision 2, CYAPCO performed an environmental review of site specific decommissioning activities. As discussed in the PSDAR Revision 2, the total occupational radiation exposure due to transportation of radioactive waste has been estimated in the PSDAR Revision 2 at approximately 54.3 person-rem. This estimate is bounded by the Supplement 1 to NUREG-0586 estimate (68 person-rem). In addition, the estimated 16.7 person-rem for public and on-lookers dose is well below the Supplement 1 to NUREG-0586 estimate (29.1 person-rem) for public and on-lookers exposure for transportation of Low Level Radioctive Waste. Radiation protection principles used during plant operation remain in effect during decommissioning to ensure that protective techniques, clothing, and breathing apparatus are used as appropriate.

11.0 REFERENCES

11.1 J. Donohew (US NRC) letter to K. Heider (CYAPCO), "Haddam Neck Plant – Issuance of Amendment Re: Approval of License Termination Plan (LTP)," dated November 25, 2002.

11.2 T. L. Williamson (Maine Yankee) letter to US NRC, "Proposed Change: Revised Activated Concrete DCGL and More Realistic Activated Concrete Dose Modeling- License Condition 2B. (10), License Termination," dated September 11, 2003.

11.3 W. Norton (CYAPCO) letter to US NRC, "Haddam Neck Plant, Revision 2 to Post Shutdown Decommissioning Activities Report (PSDAR)," dated April 28, 2004.

TABLE 1

Containment Mat Characterization Samples (Below Containment Liner)

Radio-nuclide	Sample # 175								Sample 176					
	175-4C-01	175-4C-02	175-10C-01	175-17C-01	175-20C-01	175-20C-02	175-21B-01 Bed-rock	175-21B-02 Bed-rock	176-3C-01	176-3C-02	176-8C-01	176-12C-01	176-18C-01	176-18C-02
	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	6.11	4.50	0.11	0.00	0.00	0.83	0.00	0.00	2.43	0.33	3.14	5.07	3.40	1.51
C-14	0.00	0.00							0.00	0.00				
Mn-54	0.00	0.01	0.00	0.00	0.004	0.00	0.00	0.002	0.01	0.00	0.003	0.01	0.01	0.00
Fe-55														
Co-60	0.04	0.04	0.00	0.16	0.03	0.03	0.00	0.00	0.01	0.02	0.03	0.02	0.01	0.02
Ni-63														
Sr-90	0.01	0.01	0.00	0.005	0.00	0.00	0.00	0.001	0.02	0.02	0.02	0.01	0.01	0.01
Nb-94	0.01	0.02	0.002	0.03	0.03	0.05	0.00	0.01	0.01	0.003	0.004	0.002	0.00	0.01
Tc-99														
Ag-108m	0.00	0.01	0.003	0.00	0.00	0.01	0.00	0.001	0.00	0.00	0.00	0.002	0.004	0.001
Cs-134	0.001	0.04	0.06	0.04	0.01	0.00	0.01	0.00	0.02	0.002	0.02	0.00	0.00	0.00
Cs-137	0.10	0.05	0.02	0.00	0.05	0.01	0.02	0.01	0.03	0.04	0.02	0.07	0.03	0.05
Eu-152	0.00	0.00	0.05	0.02	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.01
Eu-154	0.05	0.00	0.00	0.00	0.05	0.00	0.02	0.00	0.001	0.01	0.00	0.004	0.004	0.01
Eu-155	0.05	0.06	0.00	0.003	0.02	0.03	0.02	0.01	0.02	0.43	0.02	0.03	0.01	0.02
Pu-238	0.00	0.00							0.00	0.00				
Pu-239	0.00	0.00							0.01	0.01				
Pu-241	0.00	0.00							0.00	0.00				
Am-241	0.01	0.00	0.00	0.11	0.09	0.00	0.001	0.001	0.01	0.00	0.05	0.00	0.00	0.00
Cm-243	0.01	0.00							0.00	0.00				

Radio-nuclide	Sample 177						Sample 178					
	177-4C-01	177-4C-02	177-11C-01	177-14C-01	177-21C-01	177-21C-02	178-4C-01	178-4C-02	178-9C-01	178-13C-01	178-17C-01	178-18C-01
	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	1.01	0.92	1.13	0.25	2.82	3.95	2.34	1.41	0.00	0.41	2.22	0.73
C-14	0.00	0.00					0.43	0.00				
Mn-54	0.02	0.00	0.00	0.00	0.003	0.001	0.01	0.00	0.02	0.001	0.003	0.01
Fe-55												
Co-60	0.07	0.02	0.01	0.02	0.01	0.03	0.03	0.16	0.00	0.00	0.01	0.03
Ni-63												
Sr-90	0.02	0.00	0.02	0.004	0.02	0.12	0.01	0.02	0.00	0.01	0.00	0.01
Nb-94	0.01	0.002	0.00	0.00	0.00	0.002	0.02	0.01	0.01	0.00	0.01	0.001
Tc-99												
Ag-108m	0.02	0.001	0.001	0.01	0.001	0.001	0.001	0.00	0.003	0.01	0.005	0.01
Cs-134	0.00	0.01	0.03	0.03	0.00	0.01	0.03	0.04	0.03	0.02	0.02	0.003
Cs-137	0.03	0.03	0.02	0.02	0.01	0.01	0.04	0.02	0.00	0.02	0.05	0.58
Eu-152	0.01	0.00	0.01	0.00	0.00	0.00	0.03	0.00	0.01	0.00	0.05	0.03
Eu-154	0.01	0.02	0.00	0.005	0.00	0.02	0.00	0.02	0.04	0.001	0.00	0.00
Eu-155	0.05	0.02	0.03	0.02	0.01	0.01	0.02	0.03	0.05	0.03	0.01	0.01
Pu-238	0.00	0.00					0.02	0.00				
Pu-239	0.01	0.003					0.003	0.003				
Pu-241	2.71	2.55					1.02	0.00				
Am-241	0.00	0.01	0.00	0.00	0.00	0.00	0.01	0.01	0.02	0.10	0.00	0.05
Cm-243	0.002	0.00					0.01	0.01				

Radio-nuclide	Sample 179						Sample 180						Averages		
	179-4C-01	179-4C-02	179-9C-01	179-12C-01	179-15C-01	179-15C-02	180-4C-01	180-4C-02	180-10C-01	180-14C-01	180-18C-01	180-18C-02	Avg All Samples	Avg Inside & Out-side 1st Wafer	Avg Inside & Out-side 1st 2 Wafers H-3 Only
	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	2.35	9.34	1.70	0.18	1.76	2.52	4.48	3.52	0.65	0.00	2.24	4.78	2.056	2.753	2.389
C-14	0.00	0.00					0.00	0.00					0.036		
Mn-54	0.00	0.01	0.03	0.02	0.00	0.00	0.01	0.005	0.01	0.00	0.00	0.00	0.005	0.005	
Fe-55													N/A		
Co-60	0.02	0.00	0.01	0.01	0.003	0.02	0.01	0.03	0.00	0.01	0.00	0.01	0.024	0.027	
Ni-63													N/A		
Sr-90	0.00	0.02	0.01	0.00	0.00	0.01	0.02	0.002	0.00	0.00	0.01	0.003	0.011	0.019	
Nb-94	0.00	0.00	0.01	0.02	0.00	0.00	0.002	0.01	0.00	0.01	0.004	0.00	0.008		
Tc-99													N/A		
Ag-108m	0.0001	0.00	0.00	0.01	0.00	0.01	0.00	0.00	0.00	0.01	0.01	0.00	0.003		
Cs-134	0.06	0.03	0.08	0.00	0.01	0.03	0.02	0.00	0.03	0.03	0.00	0.02	0.019	0.016	
Cs-137	0.04	0.08	0.02	0.03	0.00	0.06	0.04	0.03	0.00	0.001	0.10	0.05	0.047	0.087	
Eu-152	0.02	0.00	0.08	0.00	0.00	0.00	0.05	0.02	0.01	0.00	0.01	0.00	0.011	0.012	
Eu-154	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.00	0.03	0.00	0.00	0.008		
Eu-155	0.02	0.06	0.09	0.01	0.02	0.04	0.05	0.02	0.03	0.05	0.02	0.03	0.037	0.027	
Pu-238	0.01	0.01					0.00	0.00					0.003		
Pu-239	0.01	0.003					0.01	0.01					0.006		
Pu-241	0.00	0.00					0.00	0.00					0.523	0.622	
Am-241	0.00	0.0005	0.11	0.08	0.01	0.005	0.03	0.01	0.00	0.00	0.00	0.00	0.018	0.009	
Cm-243	0.004	0.00					0.00	0.001					0.003		

Notes: 1. Sample Results in Bold Type are <Detectable Activity (i.e. < 2sigma error)
2. Sample Results that are less then Detectable Activity are shown as reported values (Activities < 0 are shown as 0)
3. Sample Results Underlined used in Future Groundwater Dose Calculation
4. N/A - Samples not analyzed for these radionuclides

TABLE 2

Containment Wall (Outside of Liner) Characterization Samples

	Sample # 181						Sample 182						Sample 183						Sample 184						Averages		
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 Contain-ment Wall Samples	Avg of 1st In-side and 1st Out-side	Avg 1st 2 inside and 1st 2 outside - H 3 only
	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	**0.85**	**0.53**	13.20	1.99	4.28	8.37	**0.75**	**1.15**	11.40	**0.50**	**0.87**	2.26	7.54	5.29	**0.39**	1.54	**1.11**	6.28	3.10	13.00	2.08	**0.30**	24.10	4.781	6.390	4.231
C-14	**0.00**	0.32					**0.10**	**0.00**					**0.16**	**0.00**					**0.21**	**0.07**					0.107	0.116	
Mn-54	**0.01**	**0.00**	**0.01**	**0.002**	0.02	**0.0004**	0.02	**0.00**	**0.004**	**0.00**	0.02	0.02	**0.01**	**0.01**	**0.00**	**0.00**	**0.00**	**0.00**	**0.01**	**0.00**	**0.01**	**0.01**	**0.00**	**0.01**	0.006	0.008	
Fe-55																									N/A		
Co-60	0.20	0.05	**0.01**	**0.00**	0.06	0.03	0.07	0.04	0.14	0.07	**0.01**	**0.01**	0.02	0.07	**0.00**	**0.00**	**0.00**	**0.01**	0.14	0.08	**0.02**	**0.01**	**0.01**	**0.01**	0.044	0.061	
Ni-63																									N/A		
Sr-90	0.01	0.01	0.01	0.02	**0.00**	**0.00**	**0.00**	**0.00**	**0.00**	**0.001**	**0.00**	**0.01**	0.01	**0.00**	**0.004**	**0.01**	0.00	0.03	**0.001**	0.01	0.01	0.01	**0.00**	0.01	0.006	0.008	
Nb-94	**0.002**	**0.003**	**0.00**	**0.0004**	**0.01**	**0.00**	**0.001**	**0.01**	**0.01**	**0.00**	**0.004**	**0.00**	**0.00**	**0.01**	**0.00**	**0.00**	**0.01**	**0.002**	**0.00**	**0.00**	**0.02**	**0.00**	**0.01**	**0.001**	**0.003**		
Tc-99	**0.00**	**0.00**					**0.00**	**0.00**					**0.00**	**0.00**					**0.00**	**0.00**					**0.000**		
Ag-108m	**0.00**	**0.003**	**0.01**	**0.001**	**0.00**	**0.00**	**0.01**	**0.00**	**0.001**	**0.0001**	**0.00**	**0.02**	**0.00**	**0.004**	**0.00**	**0.00**	**0.00**	**0.03**	**0.00**	**0.01**	**0.00**	**0.0003**	**0.00**	**0.01**	**0.013**		
Cs-134	**0.00**	**0.02**	**0.00**	0.02	0.04	**0.02**	**0.001**	**0.00**	**0.02**	0.02	**0.002**	**0.02**	**0.01**	**0.00**	**0.00**	**0.00**	**0.00**	**0.00**	0.03	0.03	**0.01**	**0.00**	**0.00**	0.02	0.011	0.012	
Cs-137	0.20	0.04	0.02	0.02	**0.01**	**0.00**	0.07	0.05	0.08	0.03	0.03	**0.004**	**0.01**	0.04	0.03	**0.03**	**0.01**	0.23	0.06	0.06	0.03	0.03	0.02	0.02	0.047	0.074	
Eu-152	**0.00**	**0.00**	**0.00**	**0.00**	**0.00**	**0.00**	**0.00002**	**0.00**	**0.00**	**0.03**	**0.00**	**0.04**	**0.01**	**0.001**	**0.001**	**0.00**	**0.01**	**0.00**	**0.00**	**0.02**	**0.00**	**0.00**	**0.00**	**0.01**	**0.005**		
Eu-154	**0.00**	**0.00**	**0.02**	**0.00**	**0.03**	**0.00**	**0.00**	**0.00**	**0.02**	**0.00**	**0.003**	**0.03**	**0.01**	**0.00**	**0.00**	**0.01**	**0.02**	**0.00**	**0.00**	**0.00**	**0.001**	**0.01**	**0.04**	**0.04**	**0.010**		
Eu-155	**0.05**	**0.01**	**0.02**	**0.02**	0.04	**0.04**	**0.01**	**0.0005**	**0.03**	**0.02**	0.08	**0.00**	**0.02**	**0.02**	**0.01**	0.05	**0.02**	**0.00**	**0.02**	**0.02**	**0.04**	**0.02**	**0.01**	**0.01**	0.024	0.019	
Pu-238																									N/A		
Pu-239																									N/A		
Pu-241																									N/A		
Am-241	**0.03**	**0.01**	**0.02**	**0.01**	**0.002**	**0.01**	**0.00**	**0.03**	**0.04**	**0.00**	**0.01**	**0.00**	**0.01**	**0.00**	**0.00**	**0.02**	**0.00**	**0.03**	**0.01**	**0.00**	**0.05**	**0.05**	**0.05**	**0.04**	**0.017**		
Cm-243																									N/A		

Notes:
1. **Sample Results in Bold Type are <Detectable Activity (i.e. < 2sigma error)**
2. **Sample Results that are less then Detectable Activity are shown as reported values (Activities < 0 are shown as 0)**
3. **Sample Results Underlined used in Future Groundwater Dose Calculation**
4. **N/A - Samples not analyzed for these radionuclides**

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Table 3

In Core Sump Characterization Samples

	Sample # 187								Projected Concentrations				
Radio-nuclide	187-1C-01	187-1C-02	187-1C-03	187-1C-04	187-1C-05	187-1C-06	187-1C-07	187-1C-08	Expected Average Concentration Floor (0 to 12 inch depth)	MY Ratio Upper Wall to Floor	Expected Average Concentration Upper Wall (12 inch depth)	MY Ratio Lower Wall to Floor	Expected Average Concentration Lower Wall (12 inch depth)
	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g		pCi/g		pCi/g
H-3	326.00	1590.00	2110.00	3010.00	2530.00	2890.00	2870.00	4850.00	2425.00	0.94	2.27E+03	0.72	1753.28
C-14	1600.00	5.18	3.44	3.13	2.11	1.39	0.63	0.97	0.49	0.94	4.56E-01	0.72	0.35
Mn-54	2.94	0.74	**0.01**	**0.03**	**0.00**	**0.00**	**0.00**	**0.00**	<Detection Level				
Fe-55	830.00	651.00	675.00	597.00	542.00	341.00	260.00	218.00	109.00	0.94	1.02E+02	0.72	78.81
Co-60	2230.00	392.00	319.00	258.00	245.00	216.00	127.00	98.60	49.30	0.94	4.61E+01	0.72	35.64
Ni-63	791.00	4.09	3.34	3.12	3.10	14.60	2.04	1.13	<Detection Level				
Sr-90	**N/A**								<Detection Level				
Nb-94	**0.35**	**0.42**	**0.10**	**0.13**	**0.00**	**0.18**	**0.00**	**0.00**	<Detection Level				
Tc-99	**N/A**								<Detection Level				
Ag-108m	**0.00**	**0.23**	**0.31**	**0.00**	**0.00**	**0.00**	**0.03**	**0.00**	<Detection Level				
Cs-134	6.91	5.44	4.54	3.24	2.30	1.84	1.31	0.86	0.43	0.94	4.02E-01	0.72	0.31
Cs-137	592.00	**0.00**	**0.17**	**0.12**	**0.05**	**0.20**	**0.66**	1.57	0.79	0.94	7.39E-01	0.72	0.57
Eu-152	453.00	519.00	500.00	418.00	338.00	246.00	192.00	128.00	64.00	0.94	5.99E+01	0.72	46.27
Eu-154	55.90	52.90	47.20	38.20	28.60	21.80	15.20	9.16	4.58	0.94	4.29E+00	0.72	3.31
Eu-155	**1.22**	**1.40**	2.49	**0.48**	**0.004**	**0.34**	**0.38**	**0.07**	<Detection Level				
Pu-238	0.17								<Detection Level				
Pu-239	0.17								<Detection Level				
Pu-241	**2.04**								<Detection Level				
Am-241	0.29	**0.42**	**0.00**	**0.68**	**0.35**	**0.19**	**0.14**	**0.00**	<Detection Level				
Cm-243	0.07								<Detection Level				

Notes: 1. Sample Results in Bold Type are <Detectable Activity (i.e. < 2sigma error)

2. Sample Results that are less then Detectable Activity are shown as reported values (Activities < 0 are shown as 0)

3. N/A - Samples not analyzed for these radionuclides

Table 4

Determination of Operational Building Occupancy DCGL for Inside Containment Surfaces

Sample Analysis Results - Inside Containment liner

Radionuclide	Sample 175: 175-1C-01	Sample 175: 175-1C-02	Sample 176: 176-1C-01	Sample 176: 176-1C-02	Sample 177: 177-1C-01	Sample 177: 177-1C-02	Sample 178: 178-1C-01	Sample 178: 178-1C-02	Sample 179: 179-1C-01	Sample 179: 179-1C-02	Sample 180: 180-1C-01	Sample 180: 180-1C-02
	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												
C-14	720.0	0.40	350.0	0.15								
Mn-54	0.00	0.02	0.01	0.00	0.004	0.02	0.00	0.01	0.04	0.00	0.02	0.01
Fe-55												
Co-60	7.78	0.01	23.10	0.02	6.43	0.02	3.98	0.00	5.20	0.15	1.68	0.03
Ni-63												
Sr-90												
Nb-94	0.00	0.00	0.01	0.01	0.01	0.01	0.00	0.005	0.04	0.02	0.01	0.00
Tc-99	0.00	0.00	0.00	0.00								
Ag-108m	0.00	0.01	0.01	0.004	0.00	0.00	0.00	0.01	0.02	0.00	0.01	0.01
Cs-134	0.00	0.04	0.02	0.01	0.09	0.03	0.03	0.00	0.27	0.02	0.05	0.00
Cs-137	34.90	0.02	17.00	0.06	32.50	0.06	19.70	0.09	19.50	0.10	8.98	0.02
Eu-152	0.00	0.00	0.00	0.02	0.00	0.00	0.04	0.03	0.00	0.07	0.002	0.01
Eu-154	0.02	0.00	0.06	0.001	0.02	0.003	0.07	0.01	0.07	0.04	0.05	0.00
Eu-155	0.00	0.00	0.01	0.03	0.002	0.00	0.07	0.00	0.03	0.03	0.01	0.02
Pu-238												
Pu-239												
Pu-241												
Am-241	0.17	0.04	0.26	0.04	0.03	0.00	0.03	0.00	0.03	0.00	0.00	0.04
Cm-243												

Radionuclide	Containment Sump Sample # 185: 185-1C-01	185-1C-02	185-1C-03	185-1C-04	185-1C-05	185-1C-06	185-1C-07	185-1C-08	Containment Sump Sample 186: 186-1C-01	186-1C-02	186-1C-03	186-1C-04	186-1C-05	186-1C-06	186-1C-07
	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						
C-14	70.00	0.19	0.08						25.40	0.00	0.03				
Mn-54	0.05	0.01	0.01	0.01	0.00	0.00	0.02	0.00	0.00	0.001	0.00	0.01	0.02	0.00	0.00
Fe-55	74.00								10.20						
Co-60	240.00	0.38	0.09	0.08	0.11	0.00	0.00	0.10	70.90	0.01	0.05	0.01	0.03	0.00	0.02
Ni-63	415.00								1620.00						
Sr-90	20.10								0.95						
Nb-94	0.08	0.001	0.01	0.004	0.00	0.005	0.01	0.01	0.00	0.01	0.00	0.00	0.02	0.003	0.002
Tc-99	2.84	0.00	0.00						0.33	0.00	0.00				
Ag-108m	0.00	0.003	0.00	0.00	0.00	0.0004	0.004	0.00	0.00	0.00	0.00	0.002	0.04	0.01	0.00
Cs-134	25.50	0.00	0.00	0.03	0.00	0.02	0.03	0.03	1.25	0.00	0.02	0.00	0.03	0.02	0.02
Cs-137	1270.00	6.02	0.15	0.01	0.12	0.03	0.12	3.52	584.00	1.59	0.00	0.10	0.16	0.00	0.04
Eu-152	0.00	0.15	0.08	0.06	0.07	0.06	0.02	0.00	0.00	0.00	0.00	0.00	0.03	0.00	0.00
Eu-154	1.86	0.0003	0.03	0.00	0.04	0.005	0.02	0.01	0.00	0.00	0.03	0.04	0.00	0.01	0.00
Eu-155	0.00	0.02	0.04	0.02	0.02	0.01	0.00	0.03	0.00	0.00	0.004	0.004	0.09	0.03	0.03
Pu-238	5.08								0.63						
Pu-239	1.92								0.24						
Pu-241	54.80								9.86						
Am-241	7.06	0.05	0.00	0.01	0.01	0.00	0.00	0.01	0.74	0.02	0.11	0.09	0.07	0.02	0.00
Cm-243	1.70								0.11						

Radionuclide	Average Scaling Factor (to Co-60) for Containment Floor/Sump Surface Samples	Surface Contamination Levels based on Co-60 at 1k dpm/100 cm2	Building Occupancy DCGL at 10 mrem/yr	Building Occupancy Dose based on Co-60 at 1k dpm/100 cm2	Surface Contamination Levels ratioing to 10 mrem/yr	Building Occupancy Dose after ratioing to 10 mrem/yr	% of Total Dose
		dpm/100 cm2	dpm/100 cm2	mrem/yr	dpm/100 cm2	mrem/yr	%
H-3	11.168	11168	1.26E+08	0.00	20821	1.65E-03	0.02
C-14	27.087	27087	4.12E+06	0.07	50499	0.12	1.23
Mn-54	0.003	3	1.28E+04	0.00	5	0.00	0.04
Fe-55	0.226	226	1.40E+07	0.00	422	3.02E-04	0.00
Co-60	1	1000	4.44E+03	2.25	1864	4.20	41.99
Ni-63	12.289	12289	1.44E+07	0.01	22911	0.02	0.16
Sr-90	0.049	49	5.08E+04	0.01	91	0.02	0.18
Nb-94	0.001	1	6.84E+03	0.00	3	0.00	0.04
Tc-99	0.004	4	5.80E+06	0.00	8	1.32E-05	0.00
Ag-108m	0.001	1	6.60E+03	0.00	2	0.00	0.04
Cs-134	0.028	28	6.60E+03	0.04	53	0.08	0.80
Cs-137	4.731	4731	1.72E+04	2.75	8821	5.13	51.28
Eu-152	0.001	1	9.36E+03	0.00	3	0.00	0.03
Eu-154	0.010	10	8.76E+03	0.01	18	0.02	0.21
Eu-155	0.004	4	1.75E+05	0.00	7	3.87E-04	0.00
Pu-238	0.015	15	1.95E+03	0.08	28	0.14	1.44
Pu-239	0.006	6	1.78E+03	0.03	11	0.06	0.60
Pu-241	0.184	184	9.16E+04	0.02	342	0.04	0.37
Am-241	0.012	12	1.71E+03	0.07	21	0.13	1.26
Cm-243	0.004	4	2.43E+03	0.02	8	0.03	0.33
			Totals	5.36		10.00	100.0

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

Table 5 (Page 1 of 15)
Future Groundwater Dose Calculation

A. Tritium

Volumetric Contamination

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concent-ration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	445	2.8	1225		
Containment Wall	219	6.4	1399		
SFP North Wall	16	100	1600		
SFP South Wall	16	100	1600		
SFP East Wall	12	100	1200		
SFP West Wall	12	100	1200		
SFP Floor	65	100	6500		
SFP Additional	1	100	100		
In-core Sump Top Wall	0.041	2270	93		
In-core Sump Bottom Wall	0.041	1753	72		
In-core Sump Floor	0.0168	2425	41		
		Total	**15030**	652,000	**0.58**

Surface Contamination

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	20821	1.58E-03	1.26	**913.06**	**3.50E-02**
Embedded Piping	4.45E-09	20821	9.27E-05	1.26	**53.67**	**2.06E-03**
					Total GW Dose	**6.13E-01**

Liner Surface Area Calc	Walls (2*pi*r*h)=2*pi*(67.5')*(9')=3,817 ft2			
	Floor (pi*r2) = pi*(67.5')2 = 14,314 ft2			
		Total = 18,131 ft2 or 1.68 E07 cm2		

Ci at 1 dpm/100 cm2(Liner) = (Ci/2.22 E12dpm)*(1dpm/100cm2)*1.68 E07 cm2 = 7.57 E-08 Ci

Embedded Piping surface Area = 9.8 E05 cm2 (From table 8-1)

Ci at 1 dpm/100 cm2(Embedded Piping) = (Ci/2.22 E12dpm)*(1dpm/100cm2)*9.8 E05 cm2 = 4.45 E-09 Ci

Table 5 (Cont.) (Page 2 of 15)
Future Groundwater Dose Calculation

B. Mn-54

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	9.83E-02	0.01	5.04E-04		
Containment Wall	4.84E-02	0.01	4.08E-04		
SFP North Wall	3.57E-03	0.01	3.57E-05		
SFP South Wall	3.57E-03	0.01	3.57E-05		
SFP East Wall	2.69E-03	0.01	2.69E-05		
SFP West Wall	2.69E-03	0.01	2.69E-05		
SFP Floor	1.43E-02	0.01	1.43E-04		
SFP Additional	2.53E-04	0.01	2.53E-06		
In-core Sump Top Wall	3.14E-09	0.01	3.14E-11		
In-core Sump Bottom Wall	3.14E-09	0.01	3.14E-11		
In-core Sump Floor	3.14E-09	0.01	3.14E-11		
		Total	**1.18E-03**	24200.00	**1.22E-06**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	5	3.61E-07	202	1.31E-03	1.35E-06
Embedded Piping	4.45E-09	5	2.12E-08	202	7.68E-05	7.93E-08
					Total GW Dose	**1.43E-06**

Table 5 (Cont.) (Page 3 of 15)
Future Groundwater Dose Calculation

C. Fe-55

Volumetric Contamination

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concent-ration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	8.94E-04	**<MDA**	0.00E+00		
Containment Wall	4.40E-04	**<MDA**	0.00E+00		
SFP North Wall	3.25E-05	**<MDA**	0.00E+00		
SFP South Wall	3.25E-05	**<MDA**	0.00E+00		
SFP East Wall	2.44E-05	**<MDA**	0.00E+00		
SFP West Wall	2.44E-05	**<MDA**	0.00E+00		
SFP Floor	1.30E-04	**<MDA**	0.00E+00		
SFP Additional	2.30E-06	**<MDA**	0.00E+00		
In-core Sump Top Wall	2.21E-10	102	2.25E-08		
In-core Sump Bottom Wall	2.21E-10	79	1.74E-08		
In-core Sump Floor	2.21E-10	109	2.41E-08		
		Total	**6.41E-08**	65,400	**2.45E-11**

Surface Contamination

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	422	3.19E-05	5350	4.35E-03	1.66E-06
Embedded Piping	4.45E-09	422	1.88E-06	5350	2.56E-04	9.78E-08
					Total GW Dose	**1.76E-06**

Table 5 (Cont.) (Page 4 of 15)
Future Groundwater Dose Calculation

D. Co-60

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concent-ration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	4.44E-03	0.03	1.20E-04		
Containment Wall	2.18E-03	0.06	1.34E-04		
SFP North Wall	1.61E-04	1.00	1.61E-04		
SFP South Wall	1.61E-04	1.00	1.61E-04		
SFP East Wall	1.21E-04	1.00	1.21E-04		
SFP West Wall	1.21E-04	1.00	1.21E-04		
SFP Floor	6.45E-04	1.00	6.45E-04		
SFP Additional	1.14E-05	1.00	1.14E-05		
In-core Sump Top Wall	1.23E-09	46.14	5.68E-08		
In-core Sump Bottom Wall	1.23E-09	35.64	4.38E-08		
In-core Sump Floor	1.23E-09	49.30	6.06E-08		
		Total	**1.47E-03**	**1,140**	**3.23E-05**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	1864	1.41E-04	99	1.04E+00	2.28E-02
Embedded Piping	4.45E-09	1864	8.30E-06	99	6.12E-02	1.34E-03
					Total GW Dose	**2.42E-02**

Table 5 (Cont.) (Page 5 of 15)
Future Groundwater Dose Calculation

E. Sr-90

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	5.93E-01	0.02	1.10E-02		
Containment Wall	2.92E-01	0.01	2.41E-03		
SFP North Wall	2.16E-02	0.02	4.32E-04		
SFP South Wall	2.16E-02	0.02	4.32E-04		
SFP East Wall	1.62E-02	0.02	3.24E-04		
SFP West Wall	1.62E-02	0.02	3.24E-04		
SFP Floor	8.62E-02	0.02	1.72E-03		
SFP Additional	1.53E-03	0.02	3.06E-05		
In-core Sump Top Wall	4.55E-08	0.02	9.10E-10		
In-core Sump Bottom Wall	4.55E-08	0.02	9.10E-10		
In-core Sump Floor	4.55E-08	0.02	9.10E-10		
		Total	**1.67E-02**	251	**1.66E-03**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	91	6.86E-06	45.6	1.10E-01	1.09E-02
Embedded Piping	4.45E-09	91	4.03E-07	45.6	6.45E-03	6.43E-04
					Total GW Dose	**1.16E-02**

Table 5 (Cont.) (Page 6 of 15)
Future Groundwater Dose Calculation

F. Cs-134

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	9.83E-02	0.02	1.87E-03		
Containment Wall	4.84E-02	0.01	5.83E-04		
SFP North Wall	3.57E-03	0.20	7.14E-04		
SFP South Wall	3.57E-03	0.20	7.14E-04		
SFP East Wall	2.69E-03	0.20	5.38E-04		
SFP West Wall	2.69E-03	0.20	5.38E-04		
SFP Floor	1.43E-02	0.20	2.86E-03		
SFP Additional	2.53E-04	0.20	5.06E-05		
In-core Sump Top Wall	3.14E-09	0.40	1.26E-09		
In-core Sump Bottom Wall	3.14E-09	0.31	9.76E-10		
In-core Sump Floor	3.14E-09	0.43	1.35E-09		
		Total	7.87E-03	431	4.56E-04

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	53	3.98E-06	202	1.44E-02	8.33E-04
Embedded Piping	4.45E-09	53	2.34E-07	202	8.45E-04	4.90E-05
					Total GW Dose	**8.82E-04**

Table 5 (Cont.) (Page 7 of 15)
Future Groundwater Dose Calculation

G. Cs-137

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	9.83E-02	0.09	8.53E-03		
Containment Wall	4.84E-02	0.07	3.60E-03		
SFP North Wall	3.57E-03	1.00	3.57E-03		
SFP South Wall	3.57E-03	1.00	3.57E-03		
SFP East Wall	2.69E-03	1.00	2.69E-03		
SFP West Wall	2.69E-03	1.00	2.69E-03		
SFP Floor	1.43E-02	1.00	1.43E-02		
SFP Additional	2.53E-04	1.00	2.53E-04		
In-core Sump Top Wall	3.14E-09	0.74	2.32E-09		
In-core Sump Bottom Wall	3.14E-09	0.57	1.79E-09		
In-core Sump Floor	3.14E-09	0.79	2.48E-09		
		Total	**3.92E-02**	431	**2.27E-03**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	8821	6.68E-04	202	2.41E+00	1.40E-01
Embedded Piping	4.45E-09	8821	3.93E-05	202	1.42E-01	8.23E-03
					Total GW Dose	**1.48E-01**

Table 5 (Cont.) (Page 8 of 15)
Future Groundwater Dose Calculation

H. Eu-152

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concentration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	6.94E-04	0.01	8.40E-06		
Containment Wall	3.41E-04	<MDA	0.00E+00		
SFP North Wall	2.52E-05	<MDA	0.00E+00		
SFP South Wall	2.52E-05	<MDA	0.00E+00		
SFP East Wall	1.90E-05	<MDA	0.00E+00		
SFP West Wall	1.90E-05	<MDA	0.00E+00		
SFP Floor	1.01E-04	<MDA	0.00E+00		
SFP Additional	1.79E-06	<MDA	0.00E+00		
In-core Sump Top Wall	3.84E-10	60	2.30E-08		
In-core Sump Bottom Wall	3.84E-10	46	1.78E-08		
In-core Sump Floor	3.84E-10	64	2.46E-08		
		Total	**8.47E-06**	7,330	**2.89E-08**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	<MDA	0.00E+00	3631	0.00E+00	0.00E+00
Embedded Piping	4.45E-09	<MDA	0.00E+00	3631	0.00E+00	0.00E+00
					Total GW Dose	**2.89E-08**

Table 5 (Cont.) (Page 9 of 15)
Future Groundwater Dose Calculation

I. Eu-154					
Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	6.94E-04	<MDA	0.00E+00		
Containment Wall	3.41E-04	<MDA	0.00E+00		
SFP North Wall	2.52E-05	<MDA	0.00E+00		
SFP South Wall	2.52E-05	<MDA	0.00E+00		
SFP East Wall	1.90E-05	<MDA	0.00E+00		
SFP West Wall	1.90E-05	<MDA	0.00E+00		
SFP Floor	1.01E-04	<MDA	0.00E+00		
SFP Additional	1.79E-06	<MDA	0.00E+00		
In-core Sump Top Wall	3.84E-10	4	1.65E-09		
In-core Sump Bottom Wall	3.84E-10	3	1.27E-09		
In-core Sump Floor	3.84E-10	5	1.76E-09		
		Total	**4.68E-09**	7,330	**1.59E-11**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	18	1.36E-06	3631	2.73E-04	9.33E-07
Embedded Piping	4.45E-09	18	8.00E-08	3631	1.61E-05	5.48E-08
					Total GW Dose	**9.87E-07**

Table 5 (Cont.) (Page 10 of 15)
Future Groundwater Dose Calculation

J. Eu-155

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	6.94E-04	0.04	2.60E-05		
Containment Wall	3.41E-04	0.02	8.32E-06		
SFP North Wall	2.52E-05	<MDA	0.00E+00		
SFP South Wall	2.52E-05	<MDA	0.00E+00		
SFP East Wall	1.90E-05	<MDA	0.00E+00		
SFP West Wall	1.90E-05	<MDA	0.00E+00		
SFP Floor	1.01E-04	<MDA	0.00E+00		
SFP Additional	1.79E-06	<MDA	0.00E+00		
In-core Sump Top Wall	3.84E-10	4	1.65E-09		
In-core Sump Bottom Wall	3.84E-10	3	1.27E-09		
In-core Sump Floor	3.84E-10	5	1.76E-09		
		Total	**3.43E-05**	32,500	**2.64E-08**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	7	5.12E-07	3631	1.03E-04	7.91E-08
Embedded Piping	4.45E-09	7	3.01E-08	3631	6.05E-06	4.65E-09
					Total GW Dose	**8.38E-08**

Table 5 (Cont.) (Page 11 of 15)
Future Groundwater Dose Calculation

K. Am-241

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	6.94E-04	0.02	1.27E-05		
Containment Wall	3.41E-04	<MDA	0.00E+00		
SFP North Wall	2.52E-05	<MDA	0.00E+00		
SFP South Wall	2.52E-05	<MDA	0.00E+00		
SFP East Wall	1.90E-05	<MDA	0.00E+00		
SFP West Wall	1.90E-05	<MDA	0.00E+00		
SFP Floor	1.01E-04	<MDA	0.00E+00		
SFP Additional	1.79E-06	<MDA	0.00E+00		
In-core Sump Top Wall	3.84E-10	<MDA	0.00E+00		
In-core Sump Bottom Wall	3.84E-10	<MDA	0.00E+00		
In-core Sump Floor	3.84E-10	<MDA	0.00E+00		
		Total	**1.27E-05**	13	**2.41E-05**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	21	1.62E-06	3631	3.26E-04	6.18E-04
Embedded Piping	4.45E-09	21	9.54E-08	3631	1.92E-05	3.63E-05
					Total GW Dose	**6.54E-04**

Table 5 (Cont.) (Page 12 of 15)
Future Groundwater Dose Calculation

L. Pu-241

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	9.83E-02	0.52	5.14E-02		
Containment Wall	4.84E-02	<Detectable	0.00E+00		
SFP North Wall	3.57E-03	1.00	3.57E-03		
SFP South Wall	3.57E-03	1.00	3.57E-03		
SFP East Wall	2.69E-03	1.00	2.69E-03		
SFP West Wall	2.69E-03	1.00	2.69E-03		
SFP Floor	1.43E-02	1.00	1.43E-02		
SFP Additional	2.53E-04	1.00	2.53E-04		
In-core Sump Top Wall	3.14E-09	<Detectable	0.00E+00		
In-core Sump Bottom Wall	3.14E-09	<Detectable	0.00E+00		
In-core Sump Floor	3.14E-09	<Detectable	0.00E+00		
		Total	7.85E-02	460	4.27E-03

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	342	2.59E-05	202	9.37E-02	5.09E-03
Embedded Piping	4.45E-09	342	1.52E-06	202	5.51E-03	2.99E-04
					Total GW Dose	**9.66E-03**

Table 5 (Cont.) (Page 13 of 15)
Future Groundwater Dose Calculation

M. C-14

Facility Area	Maximum Water Concentration @ 1 pCi/g Concrete Concentation	Actual Concrete Concen-tration	Maximum Water Concentration @ Actual Concrete Concentation	Ground-water DCGL @ 25 mrem/yr	Groundwater Dose at Actual Concrete Concentration
Volumetric Contamination	pCi/L	pCi/g	pCi/L	pCi/L	mrem/yr
Containment Mat	5.93E-01	<MDA	0.00E+00		
Containment Wall	2.92E-01	0.12	3.39E-02		
SFP North Wall	2.16E-02	<MDA	0.00E+00		
SFP South Wall	2.16E-02	<MDA	0.00E+00		
SFP East Wall	1.62E-02	<MDA	0.00E+00		
SFP West Wall	1.62E-02	<MDA	0.00E+00		
SFP Floor	8.62E-02	<MDA	0.00E+00		
SFP Additional	1.53E-03	<MDA	0.00E+00		
In-core Sump Top Wall	4.55E-08	0.46	2.07E-08		
In-core Sump Bottom Wall	4.55E-08	0.35	1.60E-08		
In-core Sump Floor	4.55E-08	0.49	2.22E-08		
		Total	**3.39E-02**	9,013	**9.40E-05**

Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	50499	3.82E-03	50	5.58E+01	1.55E-01
Embedded Piping	4.45E-09	50499	2.25E-04	50	3.28E+00	9.10E-03
					Total GW Dose	**1.64E-01**

N. Ni-63	**Groundwater DCGL (pCi/L)**		31500			
Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamin-ation Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	22911	1.73E-03	1891	6.69E-01	5.31E-04
Embedded Piping	4.45E-09	22911	1.02E-04	1891	3.94E-02	3.12E-05
					Total GW Dose	**5.63E-04**

Table 5 (Cont.) (Page 14 of 15)
Future Groundwater Dose Calculation

O. Tc-99	Groundwater DCGL (pCi/L)		26400			
Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamination Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	8	5.82E-07	3.27	1.30E-01	1.23E-04
Embedded Piping	4.45E-09	8	3.42E-08	3.27	7.63E-03	7.23E-06
					Total GW Dose	**1.30E-04**

P. Pu-238	Groundwater DCGL (pCi/L)		15.1			
Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamination Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	28	2.12E-06	4249	3.65E-04	6.04E-04
Embedded Piping	4.45E-09	28	1.25E-07	4249	2.14E-05	3.55E-05
					Total GW Dose	**6.39E-04**

Q. Pu-239	Groundwater DCGL (pCi/L)		13.6			
Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamination Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	11	8.05E-07	4249	1.38E-04	2.54E-04
Embedded Piping	4.45E-09	11	4.73E-08	4249	8.13E-06	1.50E-05
					Total GW Dose	**2.69E-04**

Table 5 (Cont.) (Page 15 of 15)
Future Groundwater Dose Calculation

S. Cm-243	Groundwater DCGL (pCi/L)		19.4			
Facility Area	Inventory at 1 dpm/100 cm2 (Assumes 100% Release)	Actual Surface Contamination Level	Inventory at Actual Surface Contamination level	Retardation Coefficient	Groundwater Concentration at Actual Contamination Level	Groundwater Dose at Actual Contamination levels
Surface Contamination	Ci	dpm/100cm2	Ci		pCi/L	mrem/yr
Containment Liner	7.57E-08	8	6.09E-07	30135	**1.48E-05**	**1.90E-05**
Embedded Piping	4.45E-09	8	3.58E-08	30135	**8.68E-07**	**1.12E-06**
					Total GW Dose	**2.01E-05**

Table 6-1

Future Groundwater Dose Summary

Source	Radionuclide	Future Groundwater Dose from Individual Worksheet	Percent of Dose from Concrete Sources	Percent of Dose from All Sources
		mrem/yr	%	%
Containment/	H-3	5.76E-01	9.85E+01	5.89E+01
Spent Fuel Pool	C-14	9.40E-05	1.61E-02	9.61E-03
Concrete	Mn-54	1.22E-06	2.09E-04	1.25E-04
	Fe-55	2.45E-11	4.18E-09	2.50E-09
	Co-60	3.23E-05	5.52E-03	3.30E-03
	Sr-90	1.66E-03	2.84E-01	1.70E-01
	Cs-134	4.56E-04	7.80E-02	4.67E-02
	Cs-137	2.27E-03	3.89E-01	2.32E-01
	Eu-152	2.89E-08	4.94E-06	2.95E-06
	Eu-154	1.59E-11	2.73E-09	1.63E-09
	Eu-155	2.64E-08	4.51E-06	2.70E-06
	Am-241	2.41E-05	4.12E-03	2.46E-03
	Pu-241	4.27E-03	7.29E-01	4.36E-01
	Subtotal	**5.85E-01**	1.00E+02	5.98E+01

Source	Radionuclide	Future Groundwater Dose from Individual Worksheet	Percent of Dose from Containment Liner Source	Percent of Dose from All Sources
		mrem/yr	%	%
Containment	H-3	3.50E-02	9.43E+00	3.58E+00
Liner	C-14	1.55E-01	4.17E+01	1.58E+01
	Fe-55	1.66E-06	4.48E-04	1.70E-04
	Co-60	2.28E-02	6.15E+00	2.33E+00
	Ni-63	5.31E-04	1.43E-01	5.43E-02
	Sr-90	1.09E-02	2.95E+00	1.12E+00
	Tc-99	1.23E-04	3.31E-02	1.26E-02
	Cs-134	4.56E-04	1.23E-01	4.67E-02
	Cs-137	1.40E-01	3.77E+01	1.43E+01
	Eu-154	9.33E-07	2.51E-04	9.53E-05
	Pu-238	6.04E-04	1.63E-01	6.17E-02
	Pu-239	2.54E-04	6.85E-02	2.60E-02
	Pu-241	5.09E-03	1.37E+00	5.21E-01
	Am-241	6.18E-04	1.67E-01	6.32E-02
	Cm-243	1.90E-05	5.12E-03	1.94E-03
	Subtotal	**3.71E-01**	1.00E+02	3.79E+01

Source	Radionuclide	Future Groundwater Dose from Individual Worksheet	Percent of Dose from Embedded Piping Source	Percent of Dose from All Sources
		mrem/yr	%	%
Embedded	H-3	2.06E-03	9.42E+00	2.10E-01
Piping	C-14	9.10E-03	4.17E+01	9.30E-01
	Fe-55	9.78E-08	4.48E-04	1.00E-05
	Co-60	1.34E-03	6.14E+00	1.37E-01
	Ni-63	3.12E-05	1.43E-01	3.19E-03
	Sr-90	6.43E-04	2.94E+00	6.57E-02
	Tc-99	7.23E-06	3.31E-02	7.39E-04
	Cs-134	4.90E-05	2.24E-01	5.01E-03
	Cs-137	8.23E-03	3.77E+01	8.41E-01
	Eu-154	5.48E-08	2.51E-04	5.60E-06
	Pu-238	3.55E-05	1.62E-01	3.63E-03
	Pu-239	1.50E-05	6.84E-02	1.53E-03
	Pu-241	2.99E-04	1.37E+00	3.06E-02
	Am-241	3.63E-05	1.66E-01	3.71E-03
	Cm-243	1.12E-06	5.12E-03	1.14E-04
	Subtotal	**2.18E-02**	1.00E+02	2.23E+00
Total Dose - All Sources		**9.78E-01**		1.00E+02

Table 6-2

Comparison of Concrete Activity

Current LTP to Proposed Amendment Request

Current LTP Allowable Activtiy

Radionuclide	Fraction of Future GW Dose	Dose Comp-onent of 12.5 mrem/yr	Concrete Debris DCGL	*fw* Fraction of Concrete Debris Dose from Concrete Debris	Allowable Concrete Concentra tions based on radio-nuclide mix	Total Concrete Activity for 120 Million Lbs of Concrete Debris
		mrem/yr	pci/g		pCi/g	pCi
H-3	9.83E-01	1.23E+01	90.5	0.921	48.29	2.63E+12
C-14	5.00E-05	6.25E-04	20.5	0.0135	3.80E-02	2.07E+09
Fe-55	5.60E-04	7.00E-03	89.6	0.0768	3.27E-01	1.78E+10
Co-60	5.00E-04	6.25E-03	90.7	0.0149	1.52	8.29E+10
Sr-90	2.84E-03	3.55E-02	0.377	0.107	0.01	2.73E+08
Cs-134	7.80E-04	9.75E-03	321	0.2084	0.60	3.27E+10
Cs-137	3.00E-03	3.75E-02	645	0.3537	2.74	1.49E+11
Eu-152	5.00E-04	6.25E-03	227	0.0445	1.28E+00	6.95E+10
Eu-154	5.00E-04	6.25E-03	194	0.0551	8.80E-01	4.80E+10
Am-241	1.00E-03	1.25E-02	4.42	0.7043	3.14E-03	1.71E+08
Totals	9.9267E-01	1.24E+01				

Actual Activity to Remain (Amendment)

Radionuclide	Concrete Concen-tration in Contain-ment Mat	Total Activity in Contain-ment Mat (24.7 Million lbs)	Concrete Concen-tration in Contain-ment Walls	Total Activity in Contain-ment Walls (5.3 Million lbs)	Concrete Concen-tration in In Core sump (Weighted Average)	Total Activity in Contain-ment Walls (0.5 Million lbs)	Total Activity Remaining Under LTP Amend-ment	Total Activity to Remain as a Precent of that Allowed by the Current LTP
	pCi/g	pci	pci/g	pci	pCi/g	pCi	pCi	%
H-3	2.75	3.09E+10	6.389625	1.54E+10	2091.93	4.75E+11	5.21E+11	1.98E+01
C-14	<MDA	0.00E+00	<MDA	0.00E+00	0.42	9.54E+07	9.54E+07	4.61E+00
Fe-55	<MDA	0.00E+00	<MDA	0.00E+00	94.03	2.13E+10	2.13E+10	1.20E+02
Co-60	0.03	3.02E+08	0.06	1.48E+08	42.53	9.65E+09	1.01E+10	1.22E+01
Sr-90	0.02	2.08E+08	0.01	1.99E+07	0.02	4.54E+06	2.33E+08	8.53E+01
Cs-134	0.02	0.00E+00	0.01	0.00E+00	0.37	8.42E+07	8.42E+07	2.57E-01
Cs-137	0.09	9.73E+08	0.07	1.79E+08	0.68	1.55E+08	1.31E+09	8.77E-01
Eu-152	0.01	0.00E+00	<MDA	0.00E+00	55.21	1.25E+10	1.25E+10	1.80E+01
Eu-154	<MDA	0.00E+00	<MDA	0.00E+00	3.95	0.00E+00	0.00E+00	0.00E+00
Am-241	0.02	2.06E+08	<MDA	0.00E+00	0.00	0.00E+00	2.06E+08	1.20E+02