YANKEE ATOMIC ELECTRIC COMPANY

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December 22, 2004
BYR 2004-136

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

Reference: (a) License No. DPR-3 (Docket 50-29)

Subject: Yankee Nuclear Power Station - Request for Approval of Proposed Procedures in accordance with 10 CFR 20.2002

Yankee Atomic Electric Company (YAEC) proposes to transfer certain of its solid waste from decommissioning of the Yankee Nuclear Power Station (YNPS) facilities (e.g., structures and buildings) to a disposal facility. Specifically, YAEC proposes to dispose of demolition debris from decommissioning of the YNPS facilities to the Waste Control Specialists (WCS), LLC Facility, located in Andrews, Texas. The purpose of this letter is to request NRC approval of proposed procedures for disposal of certain demolition debris in accordance with the provisions of 10 CFR 20.2002.

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

YAEC hereby requests expedited review and approval of this request by April 31, 2005 to support our decommissioning activities at the YNPS.

If you should have any questions regarding this submittal, please contact me at (413)-424-2217.

Sincerely,

YANKEE ATOMIC ELECTRIC COMPANY

J. A. Kay

J. A. Kay
Principal Licensing Engineer

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Attachments:

(1) Evaluation in Support of Alternate Waste Disposal Procedure in Accordance with 10 CFR 20.2002
(2) Waste Characterization Data (Steel, Soil & Concrete)
(3) Resident/Farmer Dose Assessment

cc: S. J. Collins, NRC Region 1 Administrator
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Docket No. 50-29
BYR 2004-136

Attachment 1

Yankee Nuclear Power Station

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

December 2004

Yankee Nuclear Power Station
Evaluation in Support of Alternate Waste Disposal Procedures
In accordance with 10 CFR 20.2002

1.0 INTRODUCTION

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

2.0 DISPOSAL SITE CHARACTERISTICS

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

2.1 Environment and Facility Design

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

The most significant natural site features that appear to limit the transport of radioactive material are the low precipitation rate and the long vertical distance to groundwater. The precipitation rate in this arid location is 0.355 meters per year (Reference 6.4 - WCS Radiological Environmental Monitoring Summary Report for 2002). The depth to groundwater accommodates a 5-meter thick cover, a 22.86 -meter thick disposal zone, and a 300-meter thick unsaturated zone between the base of the disposal cell and groundwater (Reference 6.5 Permit No. HW-50358).

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

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

Other facility design features and operating procedures provide shorter term confinement of radioactive materials and limit the potential for radiation exposure during receipt of material and emplacement of materials in the cell. WCS adheres to ALARA (Reference 6.8 WCS Radiation Safety Program).

The total capacity of the cell which would receive the YNPS waste is approximately 127,426 cubic meters. The surface area of the cell is approximately 5,574 square meters. The material that YAEC proposes for disposal if occupying the full depth of this cell would have a surface area of approximately 750 square meters. This means that the YNPS material would occupy a small percentage of the total volume of this disposal cell.

2.2 Permits

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

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

Under the State of Texas TDH general protection standards, all owners and operators disposing of radioactive materials are required to conduct operations in a manner consistent with 25 Texas Administrative Code 289.202, "Standards for Protection

Against Radiation from Radioactive Material." In addition, no owner or operator may operate in a manner such that any member of the public would receive an annual TEDE in excess of 100 millirem per year. In addition, no person may release radioactive material for unrestricted use in such a manner that the reasonable maximally exposed individual would receive an annual TEDE greater than 10 millirem per year.

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

- Waste acceptance criteria for radioactive material;

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

As previously mentioned, the analysis to follow will show that the YNPS material proposed for disposal at the WCS facility will result in doses that are a small fraction of the applicable limits.

2.3 Operations

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

Waste acceptance criteria applicable to the material intended for disposal are as follows:

- WCS Waste Acceptance Criteria (Reference 6.6)
- WCS Waste Acceptance Plan (Reference 6.7)

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

To meet these requirements, WCS conducts its operations in accordance with its Radiation Safety Program (Reference 6.8) and other operating procedures. These procedures include measures for minimizing release of material in receipt and handling. Workers use mechanized equipment to transfer and deposit material in the disposal cell. Dust suppression techniques are used daily for materials placed in the cell to minimize the potential for release of radioactive materials to the atmosphere.

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

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

The following samples are analyzed for isotopic uranium and thorium, Ra-226, gamma isotopic, Gross Alpha and Gross Beta radioactivity:

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

2.4 Post-Closure Plan

As required by the State of Texas, Texas Commission on Environmental Quality (TCEQ), WCS maintains an approved closure plan, submitted as part of its permit application (Reference 6.5). The plan conforms to all standard closure and post-closure requirements applicable to RCRA disposal facilities, including post-closure monitoring and financial assurance.

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

3.0 DESCRIPTION OF WASTE

3.1 Physical Properties

The waste material intended for disposal includes structural steel, soils associated with foundation excavations and PCB remediation, and concrete and/or pavement or other similar solid materials. The waste material proposed for disposal at the WCS facility will originate from the demolition and removal of structures and paved surfaces at the YNPS plant site, after the structure/surface has been decontaminated to remove areas that are contaminated.

The physical form of this demolition debris will be that of bulk material of various sizes ranging from the size of sand grains up to occasional monoliths with a volume of several cubic feet. YAEC, for the purpose of calculations, assumed the material to be a homogeneous mixture with a specific density of 1 gram per cubic centimeter during shipment and 1.5 grams per cubic centimeter after compaction in the disposal cell at WCS. The material will be dry solid waste containing no absorbents or chelating agents.

3.2 Estimated Waste Volume

It is estimated that the mass of demolition debris originating from the decommissioning of the YNPS will total approximately 60 million pounds. A breakdown of this waste by source is shown below in Table 3.2.1.

Table 3.2 .1

Source of Waste	Estimated Waste Weight (pounds)
Steel	5,000,000
Soil and Asphalt	15,000,000
Reactor Support Structure (RSS) Concrete	30,000,000
Other Concrete	10,000,000
Total	60,000,000

With an assumed density of 1.51 grams per cubic centimeter, (after compaction at the disposal site) the estimated volume of material to be disposed of at the WCS facility is approximately 250,000 cubic feet. This represents a small percentage of the annual volume of waste at the WCS facility. It will be conservatively assumed that all the YNPS waste material is shipped to and received at the WCS facility in one year starting in 2005. Each shipment will be assumed to at the maximum road weight of 45,000 lb. Based on a shipping capacity, obtained from the shipping contractor, of 36 shipments per week, a total waste mass of approximately 84 million pounds will be considered in this analysis.

The material will not be isolated or dedicated to a single burial cell at the WCS facility. Rather, it will be co-mingled with other radioactive and non radioactive waste material. The material will be covered at the end of each workday with an appropriate spray to lockdown contamination in accordance with WCS facility requirements.

3.3 Radiological Characterization of Waste

3.3.1 Background

YAEC is conducting extensive characterization of the remaining contaminated structures and soils on the site. The remaining structures have undergone extensive remediation over the course of the decommissioning and only low levels of residual contamination remain. The radioisotopic distributions presented in Attachment 2, Tables 1-4 represent typical radionuclide fractions for the waste form categories listed in Table 3.2.1. To ensure proper characterization of waste prior to shipment, specific and/or additional characterization data will be used as appropriate.

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

3.3.2 Characterization Results

Table 3.2.1 provides the estimated quantities of waste materials from the below sources. The information provided below represents some typical characterization data for each expected waste form. However, this radiological evaluation will result in the use of a volumetric contamination limit (pCi/g). This limit will be applied to all material that is shipped to WCS by using the Yankee truck monitoring system. This system contains 8 collimated high purity germanium detectors (HPGe) capable of detecting well below the limits established in this evaluation.

The data in Table 3.2.1 will be compared to the limits established in this evaluation to provide an estimate of the doses to members of the public from the transport and disposal of this waste. Details of the waste forms that Yankee proposes to dispose of at the WCS facility are as follows:

<u>Steel</u>

As noted in Table 3.2.1, 5 million pounds of steel is estimated for disposal at the WCS facility. The majority of the estimate consists of ~1 inch thick plate steel from the containment shell and plate steel walkways which were within the containment structure. These steel surfaces have been surveyed during dismantlement and have contamination levels in the range of 1000-2000 dpm/100sqcm. Other steel components in this category include steel beams, pipe, and framework which also have contamination levels in the

range of 1000-2000 dpm/100sqcm. Attachment 2, Table 1 presents a conservative isotopic distribution to represent this material category.

Soil and Asphalt

As noted in Table 3.2.1, 15 million pounds of soil and asphalt is estimated for disposal at the WCS facility. The soil and asphalt in this category originate from a number of areas on site and will contain low concentrations of radionuclides and may also contain PCB's. Attachment 2, Table 2 presents a conservative isotopic distribution to represent this material category.

Reactor Support Structure (RSS) Concrete

As noted in Table 3.2.1, 30 million pounds of concrete from the RSS is estimated for disposal at the WCS facility. The RSS has undergone extensive remediation and is currently ready for demolition. As a point of reference, extensive direct measurement surveys and characterization results indicate that the RSS would meet NRC Final Status Survey (FSS) unrestricted release requirements if it were to remain on site. Surveys indicate that in general, contamination levels are below 5000 dpm/100sqcm with small areas that exceed this level. Attachment 2, Table 3 presents a representative isotopic distribution that can be applied to this material category.

Other Concrete

As noted in Table 3.2.1, 10 million pounds of concrete in this material category is estimated for disposal at the WCS facility. Concrete in this material category originates from the Spent Fuel Pool (SFP), the Ion Exchange Pit (IXP), the Primary Auxiliary Building (PAB), and miscellaneous slabs. Attachment 2, Table 4 presents an isotopic distribution from the SFP that conservatively represents the concrete in this material category.

4.0 RADIOLOGICAL ASSESSMENTS

4.1 Transport Worker Dose Assessment

The Transportation Scenario Maximally Exposed Individual (MEI) dose equivalent will not exceed a few (e.g., five (5)) millirem/yr. This standard of a "few mrem/yr" to a member of the public prior to license termination is defined in NRC Regulatory Issue Summary 2004-08 (Reference 6.4). The transportation workers and worker at the WCS site are treated as members of the public as the WCS site is not licensed by the NRC. Evaluations of both internal and external dose hazards to the transportation worker are discussed below.

Each conveyance will be a strong-tight container and will be verified to be in compliance with Department of Transportation (DOT) external loose surface contamination limits

prior to shipment. Therefore, there are no internal dose hazards associated with the Transportation Scenario.

The conservative average activity concentration for each container was calculated using the penetrating gamma dose rates associated with cobalt-60 and cesium-137. The geometry model was based on a container with dimensions of 215.9 cm (7'10") length, 584.2 cm (19' 2") width, and 154.94 cm (5' 1") height with a volume of 19.5 cubic meters or 25 cubic yards. Three waste medias were used to calculate gamma exposures to the Maximally Exposed Individual (MEI), concrete rubble, soil with some moisture content, and scrap iron. Worst-case exposures were used in the resultant dose rate model, which in this case is from scrap iron. Using Microshield©, the following dose receptor points were modeled for cobalt-60 and cesium-137 separately:

- A worst case, approximately 1 meter (3 feet) receptor point adjacent center of the length or "side view" of the container for workers loading and securing the shipment;

- A typical 1.5 meter (5 feet) receptor point adjacent to the center of the length or "side view" of the container for miscellaneous worker tasks such as inspection and off loading; and

- A theoretical "driver" receptor point of 3 meters (10 feet) for transport to the TSD facility.

The resultant dose rates to each of these receptor points for cobalt-60 are 0.863 uR/hr, 0.553 uR/hr, and 0.238 uR/hr, respectively. For cesium-137, the dose rates for the same receptor points are 0.165 uR/hr, 0.106 uR/hr and 0.0455 uR/hr .

The assessment of dose to typical MEIs was modeled with a "time and motion" task analysis, some of which was provided by the WCS facillity. It is expected that 36 containers a week for 52 weeks a year will be sent to WCS for subsequent disposal. This represents 84 million pounds. Soil concentrations of 1 pCi/g for each radionuclide of concern were used to calculate exposure rates. Based upon these parameters, the calculated exposure times and rates to each MEI is provided in Table 4.1.2.

Table 4.1.2:
Estimated Exposure for Waste Transport for 1 pCi/g Co-60 and Cs-137

Maximum Exposed Individual	Hours/ year	Co-60 Max. Exposure Rate (mR/hr)	Max. Exposure (mR/yr)	Cs-137 Max. Exposure Rate (mR/hr)	Max. Exposure (mR/yr)
Driver-load truck	104	0.000863	0.0898	0.000165	0.0171
Driver-load truck	260	0.000238	0.0618	0.0000455	0.0118
Driver Total Exposure	-	-	0.152	-	0.0290
Rail yard worker	156	0.000863	0.135	0.000165	0.0257
Railroad engineer	104	0.000553	0.0575	0.000106	0.0110
Landfill driver	936	0.000553	0.517	0.000106	0.099
Landfill operator	468	0.000553	0.259	0.000106	0.0495

As demonstrated in Table 4.1.2, based on the 1 pCi/g concentration, the Maximum Exposed Individual is the "Landfill Driver" with an exposure time of 936 hours and 0.517 mR/yr dose for Co-60 and 0.099 mR/yr dose for Cs-137. It is qualitatively judged to be non-credible that the Transportation and Disposal Scenario Maximum Exposed Individual (MEI) (e.g. transportation worker, disposal facility worker or any other member of the public interacting with the transport and disposal activity) would exceed these occupancy times. Additionally, this dose assumes that the same individual provides this function for each load. Based on the personnel staffing levels at the WCS facility, this dose is expected to be divided among approximately 10 workers. However, this analysis will maintain a major portion of this conservatism since this exposure will be shared by only two workers.

Therefore, based upon the landfill driver MEI scenario, soil radionuclide concentrations that equate to a 5 mR/yr exposure is a maximum radionuclide concentration value of 20 pCi/g for Co-60, and for Cs-137, a value of 100 pCi/g. For waste containing mixtures of Co-60 and Cs-137, the sum of the fraction or unity equation would be applied, results not to exceed unity or "1". These limits will be implemented at the Yankee site for each conveyance using the truck monitor as described above.

In support of the operating permit issued by the State of Texas, WCS maintains a Radiation Protection Program including routine performance of radiation, contamination, and airborne radioactive material surveys. The facility currently conducts disposal activities involving materials similar to those described in Section 3, except that they are contaminated with source material, which has been exempted under 10CFR40. These source material isotopes (i.e., 238U and 232Th) are present in concentrations greater than, and have Derived Air Concentration (DAC) and Annual Limit on Intake (ALI) values several orders of magnitude more restrictive than the primary isotopes of concern. Despite this much larger internal dose hazard, the site has had no significant internal dose exposures. Therefore, operating experience indicates that there would be no internal dose

hazards associated with the disposal activities described herein, and on-site monitoring will be used to demonstrate and control compliance with all applicable limits

4.2 Resident/Farmer Dose Assessment

The RESRAD computer code was used to calculate the projected effect of the proposed disposal activity on future residents at the disposal site. Each of the LTP-identified radionuclides was included at a soil concentration of one (1) pCi/g, such that the resultant calculated dose equivalent to the maximum exposed individual (Resident Farmer) could be determined. A comprehensive report describing the methodology, input parameter selection, and calculation results is included as Attachment 3.

The results of the calculation show that the only radionuclide for which dose is greater than the RESRAD lower cutoff value of 1.0E-30 mrem/yr is Pu-238 indicating a maximum resident farmer dose at 1000 years post placement of 7.4E-7 mrem/pCi/g. This dose is due primarily to radon production from the decay of Pu-238 and is from the disposal of the concrete mass previously described.

Applying the isotopic distributions and results for each of the waste categories as summarized in Attachment 2, Tables 1 – 4 to the above RESRAD results, a total dose of < 2E-6 mrem/yr has been calculated. Based on this value, and a limit of between 30 and 100 pCi/g for Co-60 and Cs-137, respectively, it is extremely unlikely that the waste stream contemplated in this analysis could result in a dose that could approach the "few millirem" criteria. Therefore, this pathway need not be considered further in this dose analysis.

5.0 CONCLUSIONS

Based on the above assessment, it can be concluded that the calculated potential dose to members of the public (i.e., workers involved in the transportation to and placement of the waste and residents after closure of the site) as a consequence of the proposed waste disposal from the decommissioning activities at the YNPS at the WCS facility will be an insignificant fraction of the 25 millirem per year limit. Therefore, YAEC concludes that the proposed request for approval in accordance with 10 CFR 20.2002 will not have a significant impact on the workers, public, or the environment, and is acceptable.

6.0 REFERENCES

6.1 Texas Department of Health License Requirements

6.2 Texas Commission on Environmental Quality

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

6.4 WCS Radiological Environmental Monitoring Summary Report for 2002

6.5 WCS RCRA Permit No. HW-50358

6.6 WCS Waste Acceptance Criteria

6.7 WCS Waste Analysis Plan

6.8 WCS Radiation Safety Program

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Attachment 2

Yankee Nuclear Power Station

Waste Characterization Data
(Steel, Soil & Concrete)

December 2004

Table 1: Steel - Typical Nuclide Fraction[1]

NUCLIDE	Halflife (y)	Activity[2]
H-3	1.23E+01	n/d
C-14	5.73E+03	n/d
Fe-55	2.70E+00	33.321%
Ni-63	1.00E+02	28.905%
Sr-90	2.86E+01	0.369%
Tc-99	2.13E+05	n/d
Pu-238	8.78E+01	0.009%
Pu-239/240	2.41E+04	0.019%
Pu-241	1.44E+01	0.516%
Am-241	4.32E+02	0.024%
Cm-243/244	2.85E+01	0.027%
Co-60	5.27E+00	35.675%
Nb-94	2.03E+04	n/d
Ag-108m	1.27E+02	0.077%
Sb-Tem125	2.77E+00	n/d
Cs-134	2.06E+00	n/d
Cs-137	3.02E+01	1.058%
Eu-152	1.36E+01	n/d
Eu-154	8.80E+00	n/d
Eu-155	4.96E+00	n/d
TOTAL		100%

[1] RFIL 10 2604 SFP Composite Wipes, Analyzed by Framatome ANP Environmental Laboratory, Reference Date 6/29/04

[2] n/d = not detected. H-3 results in units of pCi/g, and H-3 concentration is independent of concentration of other nuclides. Nuclides other than H-3 results are given in percent detected activity.

Table 2: Soil - Typical Nuclide Fraction[1]

NUCLIDE	Halflife (y)	Activity[2]
H-3	1.23E+01	n/d
C-14	5.73E+03	n/d
Fe-55	2.70E+00	n/d
Ni-63	1.00E+02	n/d
Sr-90	2.86E+01	n/d
Tc-99	2.13E+05	n/d
Pu-238	8.78E+01	n/d
Pu-239/240	2.41E+04	n/d
Pu-241	1.44E+01	n/d
Am-241	4.32E+02	n/d
Cm-243/244	2.85E+01	n/d
Co-60	5.27E+00	0.065%
Nb-94	2.03E+04	n/d
Ag-108m	1.27E+02	n/d
Sb-Tem125	2.77E+00	n/d
Cs-134	2.06E+00	0.187%
Cs-137	3.02E+01	99.748%
Eu-152	1.36E+01	n/d
Eu-154	8.80E+00	n/d
Eu-155	4.96E+00	n/d
TOTAL		100%

[1] SFP-GP-03-08, Analyzed by Framatome ANP Environmental Laboratory, Reference Date 11/4/03

[2] n/d = not detected. H-3 results in units of pCi/g, and H-3 concentration is independent of concentration of other nuclides. Nuclides other than H-3 results are given in percent detected activity.

Table 3: RSS Concrete – Typical Nuclide Fraction[1]

NUCLIDE	Halflife (y)	Activity[2]
H-3	1.23E+01	198 pCi/g
C-14	5.73E+03	8.195%
Fe-55	2.70E+00	n/d
Ni-63	1.00E+02	n/d
Sr-90	2.86E+01	n/d
Tc-99	2.13E+05	n/d
Pu-238	8.78E+01	n/d
Pu-239/240	2.41E+04	n/d
Pu-241	1.44E+01	n/d
Am-241	4.32E+02	n/d
Cm-243/244	2.85E+01	n/d
Co-60	5.27E+00	17.779%
Nb-94	2.03E+04	n/d
Ag-108m	1.27E+02	n/d
Sb-Tem125	2.77E+00	n/d
Cs-134	2.06E+00	0.234%
Cs-137	3.02E+01	73.791%
Eu-152	1.36E+01	n/d
Eu-154	8.80E+00	n/d
Eu-155	4.96E+00	n/d
TOTAL		100%

[1] RSS-CB-Floor-1, Analyzed by Framatome ANP Environmental Laboratory, Reference Date 10/6/04

[2] n/d = not detected. H-3 results in units of pCi/g, and H-3 concentration is independent of concentration of other nuclides. Nuclides other than H-3 results are given in percent detected activity.

Table 4: Other Concrete - Typical Nuclide Fraction[1]

NUCLIDE	Halflife (y)	Activity[2]
H-3	1.23E+01	89.6 pCi/g
C-14	5.73E+03	0.004%
Fe-55	2.70E+00	n/d
Ni-63	1.00E+02	72.448%
Sr-90	2.86E+01	0.020%
Tc-99	2.13E+05	n/d
Pu-238	8.78E+01	n/d
Pu-239/240	2.41E+04	n/d
Pu-241	1.44E+01	n/d
Am-241	4.32E+02	n/d
Cm-243/244	2.85E+01	n/d
Co-60	5.27E+00	1.749%
Nb-94	2.03E+04	n/d
Ag-108m	1.27E+02	0.007%
Sb-Tem125	2.77E+00	n/d
Cs-134	2.06E+00	0.008%
Cs-137	3.02E+01	25.765%
Eu-152	1.36E+01	n/d
Eu-154	8.80E+00	n/d
Eu-155	4.96E+00	n/d
TOTAL		100%

[1] NFV-CB-02-05-10.5 Part-61 Results, Analyzed by Framatome ANP Environmental Laboratory, Reference Date 10/28/03

[2] n/d = not detected. H-3 results in units of pCi/g, and H-3 concentration is independent of concentration of other nuclides. Nuclides other than H-3 results are given in percent detected activity.