CIMARRON CORPORATION

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S. JESS LARSEN
VICE PRESIDENT

March 10, 1998

Mr. Kenneth L. Kalman, Project Manager
Facilities Decommissioning Section
Low Level Waste & Decommissioning Projects Branch
Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Ref: Docket No. 70-925; License No. SNM-928
Final Status Survey Report for Concrete Rubble in Sub-Area "F"

Dear Mr. Kalman:

Cimarron Corporation has recently completed Final Status Survey work on the Concrete Rubble located in Sub-Area "F". This Final Status Survey work was performed in order to demonstrate that the Concrete Rubble can be unconditionally released under the BTP Option #1 criteria (i.e., 30 pCi/g total uranium, excluding background) for soils and debris.

The purpose of this letter is to provide the above referenced report to the NRC staff for review and approval. Please find enclosed three (3) copies of this report for your review and approval. Two (2) copies of this report are for you and your staff and one (1) copy is for Mr. David Fauver. In addition, one (1) copy of this report has also been submitted to both the NRC Docket and to Mr. Louis Carson at NRC Region IV.

Please feel free to contact me if there are any questions or concerns.

Sincerely,

Jess Larsen

Jess Larsen
Vice-President

Enclosures

FINAL STATUS SURVEY REPORT FOR CONCRETE RUBBLE IN SUB-AREA F

for
Cimarron Corporation's Former
Nuclear Fuel Fabrication Facility
Crescent, Oklahoma

License Number: SNM-928
Docket No: 70-0925

Prepared for:

Cimarron Corporation
Oklahoma City, Oklahoma

March, 1998

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REFERENCE

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2. USNRC, Comments dated July 11, 1996 on the Decommissioning Plan for Cimarron Corporation.

3. USNRC, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of License for By-Product, Source, or Special Nuclear Material", August 1987.

4. USNRC, "Branch Technical Position on Disposal or On-site Storage of Residual Thorium and Uranium from Past Operations", FR. Vol. 46, No. 205, Page 52061, October 23, 1981.

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6. Cimarron Corporation Nuclear Materials License, SNM-1174, Docket No. 070-1193, terminated February 5, 1993.

7. Cimarron Corporation Letter to USNRC, August 20, 1990.

8. USNRC Letter from Mr. Richard E. Cunningham, Director , Division of Industrial and Medical Nuclear Safety to Dr. John Stauter, Director of Environmental Services, Cimarron Corporation, dated February 5, 1993.

9. Chase Environmental Group, Inc. "Decommissioning Plan for Cimarron Corporation's Former Nuclear Fuel Fabrication Facility, Crescent, Oklahoma", April 1995.

10. Chase Environmental Group, Inc. "Final Status Survey Plan for Unaffected Areas for Cimarron Corporation's Former Nuclear Fuel Fabrication Facility, Crescent, Oklahoma", October 1994.

11. USNRC Letter from Mr. Michael F. Weber, Chief Low-Level Waste and Decommissioning Project Branch, Division of Waste Management, to Mr. Jess Larsen, Vice President Kerr-McGee Corporation, dated May 1, 1995.

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13. USNRC Letter from Mr. R. A. Nelson, Acting Chief Low-Level Waste and Decommissioning Project Branch, Division of Waste Management, to Mr. Jess Larsen, Vice President, Cimarron Corporation, dated April 23, 1996.

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17. Chase Environmental Group, Inc. "Final Status Survey Plan for Phase III Area for Cimarron Corporation's Former Nuclear Fuel Fabrication Facility", Crescent, Oklahoma, June 1997.

18. USNRC, "Background as a Residual Radioactivity Criterion for Decommissioning", NUREG-1501, Draft Report, August, 1994.

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26. USNRC, "Scenarios for Assessing Potential; Doses Associated with Residual Activity", Policy Guidance PG 8-08.

27. USNRC, "Environmental Assessment Associated with the BTP Option #2 Onsite Disposal Cell at Cimarron", 1994.

TEXT

FINAL STATUS SURVEY REPORT FOR DECOMMISSIONING
CIMARRON FACILITY
CONCRETE RUBBLE IN SUB-AREA "F"

1.0 Introduction

This Final Status Survey Report is submitted by Cimarron Corporation to the Nuclear Regulatory Commission (NRC) to release for unrestricted access concrete rubble located in and near the drainage areas and the discharge area of Reservoir #2, which is located in Sub-Area "F" of the Cimarron site. Sub-Area "F" is one of the five Sub-Areas within Phase II. Sub-Area "F" is shown on Drawing No. 95MOST-RF3 (Appendix I), and includes affected and unaffected areas that have been surveyed as part of the ongoing site decommissioning process. This report provides the information and justification for leaving the concrete located in Sub-Area "F" in-place. The concrete was placed as rip-rap to correct erosion problems associated with the Sub-Area "F" drainage ways. A drawing showing the surface topography for the site is provided in Appendix I (95SITE).

This report documents the survey and sampling data obtained for the concrete and establishes the basis for unconditional release of the concrete rubble. Section 2.3 of the Cimarron Decommissioning Plan[1] provides the proposed criteria for leaving the concrete in-place. In addition, there have been several NRC comments and Cimarron responses to comments regarding concrete in drainage areas. In this report, Cimarron has incorporated NRC recommendations (see NRC Comment #18 of the "NRC Comments dated July 11, 1996 on the Decommissioning Plan for Cimarron Corporation"[2]) to consider volumetric concentration averaging as a method for unconditional release of the concrete.

The concrete rubble met all of the applicable surface contamination criteria for unconditional release when it was relocated to Sub-Area "F" drainage ways for erosion control. The concrete rubble was placed into Sub-Area "F" since decommissioning activities commenced in 1976, and was subject to various release criteria in effect which depended on the time of release. The concrete rubble originated in on-site buildings and structures undergoing decommissioning and was surveyed for alpha contamination, and in some cases for beta-gamma contamination, before it was used for erosion control in drainage areas north of Reservoir #2 and northeast of Burial Ground #1 in Sub-Area "F". However, the surface contamination release criteria in effect during the early phases of facility decommissioning were not as restrictive as those currently in place and ranged as high as 25,000 (maximum) dpm/100 cm^2 gross alpha (per Annex A to License SNM-928, Section 3.4, Revision dated August 30, 1976). In addition, practices which were approved by the NRC and in effect during the early phase of facility decommissioning did not entail surveys for beta-gamma activity prior to release when the contaminant was known or believed to be pure enriched uranium. Consequently, surveys performed more recently have identified levels of gross beta-gamma activity and gross alpha activity which exceeds the 1987 NRC unconditional release criteria contained in "Guidelines of Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct,

Source, or Special Nuclear Materials"[3]. The concrete rubble in Sub-Area "F" is representative of concrete from all areas of the buildings and structures at the facility and also contains concrete from all facility areas.

The proposed release criteria for the concrete is equivalent to the 1981 BTP[4] Option #1 criteria for soils, which is 30 pCi/g for enriched uranium. This criteria is appropriate for concrete rubble and debris due to the similarities in dose pathways. Concrete rubble is, in many respects, less likely to contribute dose to members of the public due to limited accessibility and a decreased probability of mechanical or physical dispersion. This report also demonstrates that the estimated doses due to the pathways of concern for the concrete rubble are significantly less than the 1981 BTP Option #1 limit concentrations for enriched uranium, and are insignificant in comparison to exposures resulting from natural background.

Cimarron Corporation intends to leave the concrete in place by demonstrating that the risks from the concrete remaining in place are insignificant. In addition, the safety hazards and costs associated with removal of the concrete from the drainage areas are significant. This is due to the location of the concrete in drainage areas, the random manner in which it was placed into the drainage area, and the physical hazards from rebar protrusions and unstable, irregular surfaces which could result in falling or tripping. Finally, the concrete continues to serve the intended useful purpose of preventing unnecessary erosion in the drainage and spillway areas.

This report includes a discussion of the final status survey performed to more precisely define the extent and magnitude of residual contamination present in the concrete located within Sub-Area "F". The final status survey was conducted in order to demonstrate that the guideline values for the Cimarron site have been met. The results of the Sub-Area "F" concrete rubble Final Status Survey are presented in this Report, and indicate that the estimated activity of total uranium contained in the concrete rubble is 4.6 millicuries. The maximum projected dose rate to the hypothetical resident from residual activity contained in the concrete rubble was calculated to be approximately one millirem per year based upon the RESRAD computer code.

2.0 Background

Cimarron Corporation, a subsidiary of Kerr-McGee Corporation, operated two plants near Crescent, Oklahoma, for the manufacture of enriched uranium and mixed oxide reactor fuels. The 840-acre Cimarron site was originally licensed under two separate SNM Licenses. License SNM-928[5] was issued in 1965 for the Uranium Plant (U-Plant) and License SNM-1174[6] was issued in 1970 for the Mixed Oxide Fuel Fabrication (MOFF) Facility. Both facilities operated through 1975, at which time they were shut down and decommissioning work was initiated.

Decommissioning efforts at the MOFF Facility were completed in 1990 and Cimarron Corporation applied to the NRC on August 20, 1990[7], to terminate License SNM-1174. After confirmatory surveys, the NRC terminated the MOFF Facility License, SNM-1174, on February 5, 1993[8].

Decommissioning efforts involving characterization, decontamination and decommissioning for the 840-acres, licensed under SNM-928, were initiated in 1976 and are still ongoing. The final objective of the decommissioning effort is to release the entire 840-acre site for unrestricted use.

Based upon historic knowledge of site operations and the characterization work completed to date, Cimarron Corporation completed and submitted in October 1994 the Cimarron Radiological Characterization Report.[9] As discussed in this report, the site has been divided into affected and unaffected areas. The affected and unaffected areas are shown on Drawing No. 95MOST-RF3, included in Appendix I. For the Final Status Survey Plan, the entire 840-acre site has been divided into three major areas which contain both affected and unaffected areas. Each of these three major areas are also shown on Drawing No. 95MOST-RF3 and are designated by Roman Numerals I, II, and III (herein referenced as Phases I, II, and III). These three major areas are then further subdivided into smaller Sub-Areas (i.e. A, B, C, D, etc.).

2.1 Phase I Area

As presented in the Cimarron Decommissioning Plan,[1] the Final Status Survey Plan (Phases I, II and III) was discussed in general terms, with the understanding that each of the three phases would be submitted to the NRC under separate cover for approval. The Final Status Survey Plan for the first of these three phases (Phase I[10]) was approved by the NRC via letter dated May 1, 1995.[11] The Final Status Survey Report[12] for Phase I was submitted to the NRC and confirmatory sampling for the Phase I areas has been completed by the Oak Ridge Institute for Science and Education (ORISE). Cimarron Corporation received license Amendment #13 from the NRC to release this area from SNM-928; the amendment was forwarded by letter dated April 23, 1996[13]. This amendment reduced the licensed facility acreage from 840 to 152 acres.

2.2 Phase II Area

The area designated as Phase II on Drawing No. 95MOST-RF3 (Appendix I) contains both affected and some contiguous unaffected areas, and represents approximately 122 of the remaining licensed 152 acres. The Final Status Survey Plan for Phase II was submitted to the NRC in July 1995[14] and approved by the NRC on March 14, 1997[15]. Phase II includes Sub-Areas F, G, H, I and J. Included within Phase II are Burial Area #1 which was released in December 1992 by the NRC[16], backfilled with clean soil, and seeded. Also included in Phase II are the East and West Sanitary Lagoons, the MOFF Plant Building exterior and yard area, the Emergency Building, the Warehouse Building (Building #4) and surrounding yard, and numerous drainage areas. Cimarron has substantially completed the remediation of each Sub-Area and final status surveys are currently underway. In general, Sub-Area "F" is located north of Reservoir #2 and includes the roadway from the northern end of Reservoir #3 to the northern end of Reservoir #2. The concrete rubble within Sub-Area "F" is located on the berm area and in the drainage area to the north of Reservoir #2 and also alongside a drainage to the

northeast of Burial Area #1. The concrete rubble addressed in this report includes an area of approximately 0.3 acres of the entire 17 acre area within Sub-Area "F". The final status survey for the remaining acreage has been completed and is being assembled for submission under a separate report.

2.3 Phase III Area

The Phase III area survey is the last phase for completing the final status survey for the entire Cimarron site, and represents approximately 30 acres. This area is designated as Phase III on Drawing No. 95MOST-RF3. The Final Status Survey Plan for release of this area from the site license, has been submitted to the NRC[17] for approval. The Phase III area includes the Uranium Processing buildings and yard area, Burial Areas #2 and #3, the New Sanitary Lagoon, the New On-site Disposal Cell (Burial Area #4), and the Five Former Waste Water Ponds. These five former waste water ponds consist of Uranium Waste Ponds #1 and #2, the Plutonium Waste Pond, the Uranium Emergency Pond, and the Plutonium Emergency Pond.

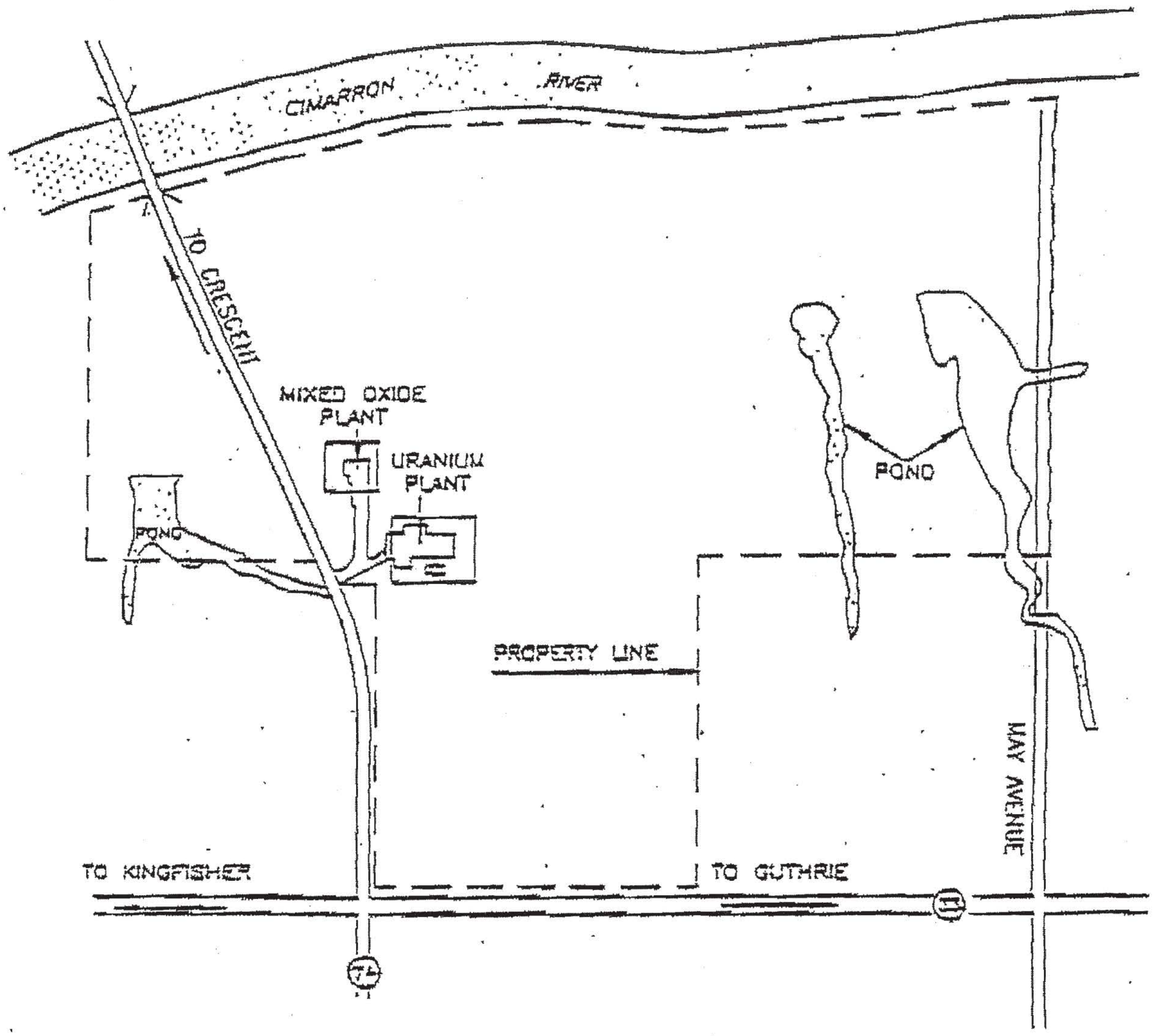
3.0 Site Description

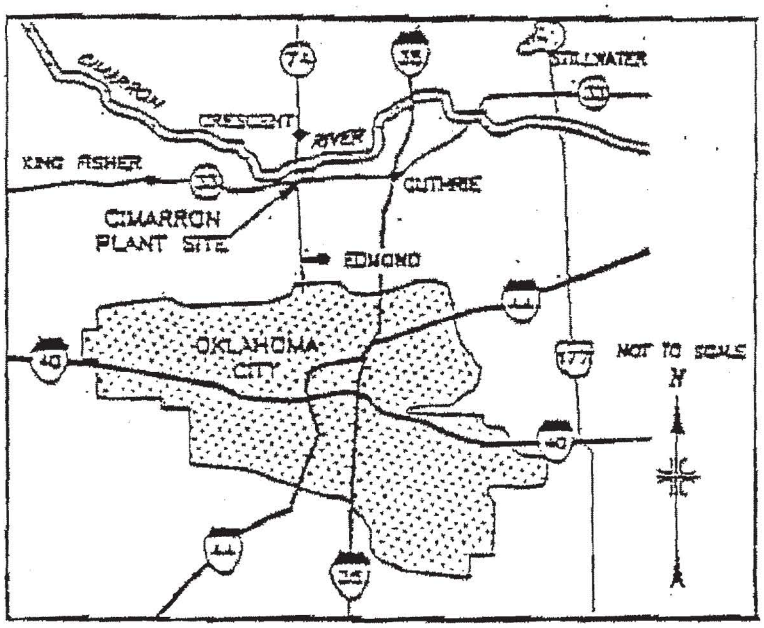
The Cimarron Facility is located in Logan County, State of Oklahoma, on the south side of the Cimarron River approximately 0.5 miles north of the intersection of Oklahoma State Highways #33 and #74. Figure 3.1 shows the site location. The 840-acre site (the licensed portion was recently reduced to 152 acres) is located in an area of low, rolling hills and incised drainages. Local elevations range from about 940 feet along the river to 1,010 feet Mean Sea Level at the plant. The county is primarily rural with an economy primarily based upon agriculture and ranching. The entire site is owned by Cimarron Corporation, a wholly owned subsidiary of Kerr-McGee Corporation.

In general, Sub-Area "F" is located north of Reservoir #2 and includes the roadway from the northern end of Reservoir #3 to the northern end of Reservoir #2. The concrete rubble within Sub-Area "F" is located on the berm area and in the drainage area to the north of Reservoir #2 and also alongside the drainage to the northeast of Burial Area #1.

4.0 Facility Description

License SNM-928 was originally issued in 1965 to Kerr-McGee Nuclear Corporation for the manufacture of enriched uranium reactor fuels. The Uranium Plant (U-Plant) was constructed to be a complete nuclear fuel service facility. Initial equipment provided for the production of UO_2, UF_4, uranium metal and the recovery of scrap materials from facility operations. In 1968 the plant was expanded by increasing the UO_2 and Pellet facilities through the installation of another complete production line for the fabrication of fuel pellets. In 1969 fabrication facilities were added for the production of fuel pins. In 1970 facilities were added for the production of the fuel





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Cimarron Corporation
Crescent Oklahoma Facility
Location Map

FIGURE 3.1

elements. Equipment initially installed for the recovery of enriched scrap material was not used after work performed under a scrap recovery contract was completed in 1970. All equipment utilized in fuel production activities has been either decontaminated and removed from the site for salvage or packaged and transported offsite for disposal at a commercial LLRW facility.

5.0 History of Site Operations

The Cimarron Facility was originally licensed under two separate licenses. License SNM-928 was issued for the U-Plant Facility and License SNM-1174 was issued for the MOFF Facility. Both facilities terminated production operations in 1975. Decontamination and decommissioning of the MOFF Facility was completed by 1990, and the license was terminated by the NRC in 1993. The U-Plant Facility decommissioning is in progress and nearing completion. A complete history of site operations can be found in both the Characterization Report[9] and the Decommissioning Plan[1].

6.0 Concrete Rubble Decommissioning Activities

The purpose of this section is to discuss the status of the on-going site decommissioning activities related to the concrete rubble located in Sub-Area "F" and to present the radiological criteria and guideline values utilized throughout the remediation and final status survey.

6.1 Identification of Contaminants

Based upon the knowledge of past site operations, the results of numerous characterization efforts to date, and other independent characterization efforts by regulatory agencies and their respective subcontractors, the possible radiological contaminants within this Sub-Area have been determined to consist of U-234, U-235 and U-238. The uranium is comprised of enriched forms, with an average enrichment above the naturally occurring level. The average U-235 enrichment at Cimarron has been previously established as approximately 2.7 weight percent.

Thorium, although not considered a contaminant of concern for the concrete rubble, has been included in the soil and sediment analysis for comparison with background levels found in other areas of the facility. Thorium was not used or processed at the facility.

6.2 Site Background Levels

6.2.1 Natural Background Radioactivity of Concrete

Concrete contains naturally occurring elements from the earth which emit radiation. NUREG-1501, "Background as a Residual Radioactivity Criterion for Decommissioning"[18], gives the typical radionuclide content for concrete used in the

United States. Table 2.3 of NUREG-1501 gives the range of U-238 as 19-89 Bq/kg (0.5-2.4 pCi/g), as reported by Eichholz[19]. Assuming radiological equilibrium between the Uranium-238 and it's radioactive daughter products through Ra-226, as suggested in NUREG-1501, the total uranium concentration (i.e., U-234 + U-235 + U-238) would range from 38-178 Bq/kg (1.0 to 4.8 pCi/g). In addition to uranium, Table 2.3 of NUREG-1501 states that concrete contains Th-232 (15 to 118 Bq/kg or 0.4 to 3.2 pCi/g) and K-40 (262 to 1147 Bq/kg or 7.1 to 31 pCi/g).

Ingersoll[20] collected ordinary concrete samples from around the country. These data are presented in Table 2.7 of NUREG-1501 and show a range for U-238 of 8 to 38 Bq/kg (0.2 to 1.0 pCi/g). Assuming radiological equilibrium between the U-238 and it's radioactive daughter products through Ra-226, the total uranium concentration (i.e., U-234 + U-235 + U-238) would range from 16-76 Bq/kg (0.4 to 2.1 pCi/g). Ingersoll also measured similar variations in concentrations of Th-232 and K-40.

A representative sample of "background" concrete was collected from the unaffected floor area under the former facility manager's office. This sample is representative of "background" concrete due to the fact that it: 1) Was poured at about the same time as the concrete rubble in Sub-Area "F"; 2) Was located in an office area of the facility that was never contaminated; and, 3) Had been covered with linoleum tile prior to use of the facility for uranium processing. This background sample was analyzed by an outside independent laboratory and was reported as containing 1.1 $\pm$ 0.4 pCi/g U-234, $<$0.2 pCi/g U-235, and 0.4 $\pm$ 0.2 pCi/g U-238. The sample was also analyzed using the on-site soil counter which produced a total uranium concentration of 5.7 $\pm$ 1.3 pCi/g. Conversion of the onsite counter results to natural uranium, using the conversion factor of 0.67/1.5 as described in Section 6.2.3, results in a total uranium concentration of 2.5 pCi/g. This confirms that this concrete sample is similar in concentration to background soils and is within the range of activities listed in NUREG-1501 (after consideration of measurement uncertainty).

Concrete, such as that in the Sub-Area "F" drainage areas, contains naturally occurring radioactivity such as Uranium, Thorium and Potassium-40 which affect instrument measurements and laboratory analysis. The natural background concentrations from activity due to total uranium in concrete are therefore subtracted from counting results to obtain the contribution of activity which resulted from facility operations. The total uranium background for concrete which will be used in this report is 1.5 pCi/g, based upon the sample collected at the facility and the literature which has been cited.

6.2.2 Establishment of a Background Value for Gross Alpha and Gross Beta-Gamma Surface Activity in Concrete

The average background for gross beta-gamma surface activity in concrete was determined using data gathered during the survey of the concrete rubble as well as data from the representative sample of "background" concrete as described in Section 6.2.1.

In order for data to qualify as representative of background, data from each selected 5m x 5m grid area had to have gross beta-gamma levels (average) of less than two times background (for concrete). Grid areas were initially selected by utilizing the data from areas with gross alpha survey results of less than 100 dpm/100cm^2 (maximum). In addition, the reported exposure rate at both the surface and at a distance of one meter for the grid area could not exceed 10 µR/h and could not be greater than 1 µR/h above the "one meter" measurement location at any point within the grid area. Data utilized in the determination of gross alpha and gross beta-gamma surface activity background are summarized in Table 6.1.

Table 6.1
Gross Alpha and Gross Beta Surface Activity Background Data

Location	Ave. Alpha Dpm/100cm^2	Max. Alpha dpm/100cm^2	Ave. Beta dpm/100cm^2	Max. Beta dpm/100cm^2	Surface µR/h**	1m µR/h**
Grid #6	10	40	1045	1650	7	8
Grid #7	10	40	278	1584	7	6
Grid #8	10	40	924	1397	7	7
Grid #9	10	80	557	935	6	6
Grid #21	50	80	1199	3212	6	6
Grid #33	5	20	889	1386	7	7
Grid #103	10	40	860	1353	7	7
Representative "background"	60*	60*	649*	649*	5*	5*
Average	**21**	**50**	**800**	**1520**	**6.5**	**6.5**

*Core sample-measurements taken at the surface of the area to be cored prior to sampling. As this area is within a building over a concrete slab, the ambient background exposure rate was lower.

**Exposure rate measurements include background contribution. All other measurements are net (instrument background subtracted).

Based upon Table 6.1, the average background for gross alpha surface activity was established as 21 dpm/100 cm^2. The gross alpha background activity is presented for reference purposes only. The data and evaluations presented in this report did not utilize the gross alpha background for subtraction as the value was insignificant to the overall conclusions made in the report. Thus, the data tables and drawings reflect the gross alpha activity, including background.

The average background for gross beta surface activity was established at 800 dpm/100 cm^2. The gross beta background was subtracted from the surface activity data prior to calculation of the volumetric activity concentration of total uranium.

6.2.3 Soils

Natural background levels for uranium in soil have been established through numerous measurements by Cimarron personnel utilizing the on-site soil counter and through

independent laboratory analysis. Analytical results from Cimarron Corporation's environmental sampling program are reported to the NRC annually in Environmental Reports. These reports provide sample analysis results for soil samples collected from numerous off-site locations which are representative of background in surrounding soils.

Cimarron personnel collected and analyzed 30 surface soil samples from the perimeter of the Cimarron site during the first quarter of 1995 to further validate background levels. Total uranium ranged from 2.3 pCi/g to 6.6 pCi/g, with the average being 4.0 ± 2.6 (2σ) pCi/g. These values were obtained as a result of using the Cimarron on-site soil counter. The on-site soil counter is calibrated to assume an enrichment of 2.7 weight percent as this is the average enrichment found throughout the site. When a correction factor (0.67/1.5) is applied to these results to convert the values from an assumed 2.7 weight percent enrichment to a natural enrichment, the converted results ranged from 1.0 pCi/g to 2.9 pCi/g with an average of 1.8 ± 1.0 (2σ) pCi/g total uranium. Based upon these results, the average value of 4 pCi/g total uranium for background was used when the soil sample analytical results were compared to guideline values.

6.2.4 Exposure Rates

Background exposure rates have been established at the Cimarron site by taking micro-R readings and pressurized ion chamber (PIC) readings at off-site sample locations in addition to Cimarron site areas which are unaffected by past operations. Site background exposure rates of approximately 7-10 μR/h have been observed in background areas by Cimarron personnel utilizing a Ludlum Micro-R survey meter, and have been used in past reports to the NRC. Site background exposure rates of approximately 7-10 μR/h have also been determined by ORISE personnel utilizing similar instrumentation. In addition, site background exposure rates were measured by ORAU (now ORISE) personnel utilizing a PIC[21], and were determined to be 9 to 10 μR/h. Thus, depending upon instrumentation utilized, the background exposure rate at the Cimarron site ranges from 7 to 10 μR/h.

Cimarron personnel performed exposure rates measurements at background locations in Sub-Area "F" in 1995 using a Micro-R meter. Confirmatory measurements were obtained at the same locations in 1997 using a Reuter-Stokes PIC. These data are tabulated below in Table 6.1. The average background as measured using the Micro-R meter was 7.6 ± 1.3 (2σ) μR/h, and is about 15 percent less than the average for the PIC measurements of 9.0 ± 1.1 (2σ) μR/h. These differences are not significant and indicate good agreement between the Micro-R measurements and the PIC measurements. Cimarron will continue the use of 7-10 μR/h as representative of background exposure rates for Micro-R measurements in accordance with past reports.

TABLE 6.2			
Sample ID No.	**Grid Location**	**Micro-R Reading (μR/h)**	**PIC Reading (μR/h)**
UAF-BKG-1	819W-81N	9	9.8
UAF-BKG-7	1600E-120N	7	7.6
UAF-BKG-11	840W-700S	8	9.5
UAF-BKG-13	840W-288S	9	9.8
UAF-BKG-16	808W-282S	8	9.7
UAF-BKG-19	640W-700S	9	10.5
UAF-BKG-23	1610E-300S	5	7.8
UAF-BKG-25	1610E-69N	6	7.6
UAF-BKG-27	1610E-469N	7	7.8
UAF-BKG-28	1610E-634N	8	9.6
	AVERAGE	**7.6 ± 2.7 (2σ)**	**9.0 ± 2.3 (2σ)**

6.3 Characterization Data

As noted earlier, the Cimarron site has been subdivided into survey units. These units are naturally distinguishable or have a common history of use, characterization and decommissioning activities. Throughout most of the decommissioning process at the Cimarron site, a unit was characterized, remediated (if required), and resurveyed. The description of the decommissioning activities and the final status survey data were then submitted to the NRC for review and approval. After review of the submittal, the NRC either released the unit and/or contracted with ORISE (previously ORAU) to perform a confirmatory survey. Based upon the ORISE confirmatory survey (if requested by the NRC), the NRC either would release the unit or require additional remediation. The concrete rubble within Sub-Area "F" is one such unit. Cimarron personnel have completed the characterization and final status survey of the concrete rubble.

6.4 Environmental Monitoring Data

The concrete rubble in Sub-Area "F" is near one location where environmental surface water monitoring has been performed in accordance with the Cimarron Environmental Sampling Program. All analyses for samples collected from this location are performed by an independent off-site laboratory.

The surface water monitoring at this location (location #1205) consists of an annual sample from Reservoir #2, which is upstream from the concrete rubble. The location of surface water sampling location #1205 is shown in Drawing No. 95MOST-RF1 (Appendix I). Since 1986, Gross α concentrations at this location have been reported as less than 10 pCi/l, and gross β concentrations have been less than 20 pCi/l. Total U has

been reported as less than 0.005 mg/l over the same period. The laboratory reported 0.5 pCi/l U-234, <0.1 pCi/l U-235, and 0.2 pCi/l U-238 for surface water location #1205 in 1997. Based upon the historical environmental data, Reservoir #2 has not been affected by facility operations or decommissioning activities.

The final status survey for the concrete rubble also included the collection and analysis of additional water samples from the stream originating from Reservoir #2. These results are discussed in Section 8.3 of this Report.

7.0 Final Status Survey Procedure

The purpose of this section is to describe the methodology utilized for the collection of the Final Status Survey data. The final status survey data will be used to demonstrate that the applicable radiological parameters (i.e., guideline values) have been met and that the concrete rubble in Sub-Area "F" can be released from License SNM-928. Due to the nature and location of the material and the physical limitations involved with the performance of surveys, Cimarron Corporation developed innovative calculational techniques to demonstrate compliance with the release criteria.

7.1 Survey Procedure

A 5m x 5m grid system was established for the concrete rubble. The grid system was established such that each grid area could be easily relocated in the future for additional survey work and/or confirmatory surveys. The concrete grid system can be described as consisting of three areas, consisting of: 1) the main body of concrete in the drainage area north of the reservoir (Grid #'s 20 through 122); 2) the concrete along the west bank of the spillway area (Grid #'s 1 through 19); and, 3) the concrete west of the main drainage area, northeast of Burial Ground #1 (Grid #'s 123 through 134). Grid locations are shown on Drawing No. 98FCONC-0.

The concrete was then further subdivided into those grids containing ≥85% concrete surface area and grid areas containing significantly less concrete. A computer generated random sampling plan representing 32 of the grids containing significant areas of concrete was developed. The random sample plan also contained those accessible areas known or suspected to have the highest gross beta-gamma surface contamination. The random sample plan thus should represent a worst case average for the concrete. The random sample represents 24% (32 of the 134) 5m x 5m grid areas containing concrete rubble. However, since the random sample was selected from grid areas containing >85% concrete, the random sample should generate data which represents even a higher percentage of the surface area for the concrete rubble as a whole. The grid areas representing the random sample are also shown in Drawing No. 98FCONC-0 (Appendix I).

Each randomly selected 5m x 5m grid area was then 100% scanned, to the extent practicable, for gross beta-gamma activity to identify any elevated areas of activity. Any location within each grid that exceeded 5,000 dpm/100cm^2 was documented on survey forms. The 5,000 dpm/100cm^2 cut-off value was chosen because it represents the unconditional release surface activity criteria (average) for uranium. Supplemental gross beta-gamma measurements were also collected for each of the 5m x 5m grid areas containing concrete along the west bank of the spillway, as well as at other locations, as shown in Drawing 98CONC-0 (Appendix I).

Gross alpha scans to identify elevated locations were not performed due to the fact that the concrete had been previously surveyed and released (all gross alpha was assumed to be due to uranium). Additionally, gross alpha scanning over rough surfaces has not proven to be an effective method due to the effects of geometry (source to detector distance) and attenuation (shields the alpha particles before they are able to reach the detector). The survey data for areas containing elevated measurements of gross beta-gamma activity versus the gross alpha measurement results provides evidence of this (See Tables 3 through 6 in Appendix II). It was therefore determined that the most effective method for locating the elevated areas was through the use of gross beta-gamma scans.

A 1m x 1m grid was established surrounding any location within a selected 5m x 5m grid area which exceeded 5,000 dpm/100cm^2 beta-gamma (i.e., hot-spot areas). Surveys or scans were then performed in each 1m x 1m grid area as described below.

- The average gross beta-gamma dpm/100cm^2 was measured.
- The maximum gross beta-gamma dpm/100cm^2 was measured.
- The average gross alpha dpm/100cm^2 was measured.
- The maximum gross alpha dpm/100cm^2 was measured.
- A μR/hour reading at the surface and at one meter above the surface directly above the location with the highest beta-gamma activity was measured.

In addition, a "representative" 1m x 1m survey area was selected from each of the selected 5m x 5m grid areas. The purpose for the representative area was to establish survey data that represented the surface areas within each 5m x 5m grid that did not contain any locations with elevated surface activity (i.e., above the 5,000 dpm/100cm^2 "hot-spot" criteria). The methodology utilized in locating these representative areas was to locate them in areas that were representative of the remainder of the concrete surfaces within the 5m x 5m grid area, preferably at the center of the grid area, but not in an area previously determined to have elevated readings above 5,000 dpm/100cm^2 beta-gamma. Measurements were performed in each of the representative grid areas as described above for the "hot-spot" areas exceeding 5,000 dpm/100 cm^2.

In addition to the 32 random survey measurements, Cimarron personnel performed final status surveys on 34 grid areas to supplement the random sample data and to provide assurance that the random sample survey data were representative. The

supplemental surveys were performed in exactly the same manner as the random sample surveys.

Surface activity measurements for gross beta-gamma were obtained using a Ludlum Model 2221 with a Model 43-68 gas proportional probe, or equivalent (e.g., Ludlum Model 43-89). Gross alpha surface scans were obtained using an Eberline Model PRM-6 with an Eberline Model AS-15 probe. Micro-R measurements were obtained utilizing a Ludlum Model 19. The top surface and any other accessible concrete surfaces were surveyed. This intent of the survey was to be as representative of the concrete rubble as a whole, through the use of statistically based sampling techniques and the application of reasonable calculational assumptions.

Removable contamination measurements were not obtained due to the fact that the concrete has been in the drainage area for over 10 years. It was determined that any potentially removable contamination remaining would have been removed by environmental interactions. Monitoring of soils and sediments downstream of the concrete provided confirmation that removable contamination has not contributed to natural concentrations of uranium in downstream soils and sediment.

7.2 Exposure Rate Measurements

Exposure rate measurements were obtained at the surface of the concrete and at one meter above the concrete surface utilizing a sodium iodide based micro-R meter, or equivalent. These measurements support the position that the concrete rubble does not pose any significant external dose hazard and that exposures are similar to background.

7.3 Environmental Exposure Rate Measurements

Three environmental thermoluminescent dosimeters (TLDs) were placed near and above the concrete rubble to support the results of previous exposure rate measurements and to demonstrate that there are not any seasonal or long-term upward trends in exposure rates. Data from these TLDs are compared to site background area TLD measurements.

7.4 Surface Water Sampling

Two surface water samples were collected from areas where surface water is normally available. These surface water samples were collected up-gradient and down-gradient from the concrete rubble area. The water samples were analyzed for isotopic uranium and thorium at an independent off-site analytical laboratory.

7.5 Soil/Sediment Sampling

Thirteen soil/sediment samples were collected from the concrete rubble area to demonstrate that the concrete is not contributing to levels of uranium and thorium

present in the sediment. One soil/sediment sample was collected up-gradient of the concrete, two samples immediately down-gradient in locations likely to collect sediments, and ten samples from soils and sediments beneath or within the concrete rubble. All samples were analyzed for total uranium and thorium activity utilizing the Cimarron soil counter. One sample collected down-gradient from the concrete rubble was also sent to an independent off-site laboratory for analysis.

7.6 Relationship Between Surface Activity and Concentration

A special study was conducted to determine the depth profile of the contamination in the concrete and for the purpose of establishing a relationship between gross beta-gamma surface activity (measured in dpm/100cm^2) and uranium concentrations (measured in pCi/g). A brief outline of this study is provided below:

1. Concrete rubble slabs with the following approximate gross beta-gamma surface activity were selected:

   - Background ($\leq$1,000 dpm/100cm^2 average beta-gamma)
   - $\approx$ 5,000 dpm/100cm^2 average beta-gamma
   - $\approx$ 20,000 dpm/100cm^2 average beta-gamma

2. The following measurements were then performed on each piece of concrete rubble selected for the study;

   - Exposure rates at the surface and at 1 meter above the surface utilizing micro-R meter;
   - Gross alpha dpm/100cm^2 - one measurement for each probe sized area located inside the area to be scabbled;
   - Gross beta-gamma dpm/100cm^2 - one measurement for each probe sized area located inside the area to be scabbled;
   - Hot-spot - all of the measurements described in #2 performed at the hot-spot.

3. The thickness of the concrete slab was measured.

4. The elevated area on the concrete rubble was scabbled to remove a layer of residual contamination.

5. Concrete scabble dust was collected and mixed thoroughly. A sample was obtained and analyzed using the on-site Soil Counter and/or an independent off-site laboratory.

6. All measurements were recorded, including the estimated thickness of the scabbled area.

7. Steps #2 through #6 were repeated until measurements of the scabbled area indicated that gross beta-gamma activity was reduced to less than twice background.

7.7 Guideline Values

The radiological guideline values discussed in this section are utilized for comparison to the final survey data to verify that the concrete rubble in Sub-Area "F" can be released from License SNM-928.

7.7.1 Concentration Guidelines for Concrete

Draft NUREG/CR-5849[22], Section 2.2, states that "Volume concentration guideline values, which apply to soil, induced activity, and debris, are expressed in terms of activity per unit mass [typically picocuries per gram (pCi/g).]". Cimarron Corporation has established the release guideline values for concrete in Sub-Area "F" in accordance with the Branch Technical Position[24] (BTP) Option #1. The BTP Option #1 criteria is 30 pCi/g total uranium (enriched), above background.

Due to the physical obstacles and potential safety hazards associated with monitoring the concrete rubble, it was necessary to develop new calculational methods for determining volumetric concentrations. The overall objective of the survey effort was to demonstrate that the release guideline values were complied with. Due to the nature of the random sampling performed, the methods of draft NUREG/CR-5849 for "hot-spot" averaging were not directly utilized. Rather, justification that the overall average concentration meets the release criteria (i.e., BTP Option #1 guideline value) is provided herein. This was determined to be appropriate based upon the characteristics of the concrete rubble and the low probability that a portion of the concrete rubble would be extracted and used in a manner that could contribute significantly to inadvertent exposure.

7.7.2 Exposure Rate Guidelines (External Dose)

The exposure rate (external dose) guideline value was established as 10 micro-Roentgens (μR) per hour (average) above background at one meter above the surface in accordance with the BTP[4] Option #1 criteria. As stated in the BTP, this is compatible with proposed EPA cleanup standards for inactive uranium processing sites.

Exposure rates may be averaged over a 100 m^2 grid area as described in draft NUREG/CR-5849[22]. Draft NUREG/CR-5849 also states that the maximum

exposure rate at any discrete location within a 100 square meter area cannot exceed 20 μR/h above background. Cimarron Corporation utilizes 7 to 10 μR/h as the average background exposure rate.

7.7.3 Volumetric Activity of Soils and Sediments

The guideline value for residual concentrations of total uranium which may remain in soils or sediments is specified as BTP Option #1 material for Sub-Area F. For enriched uranium, as specified in Table 2 of the BTP[16], the Option #1 limit is 30 pCi/g total uranium above background. The amount of soil and sediment sampling performed during this final status survey was confined to selected locations which were utilized to demonstrate that the concrete had not resulted in significant impact to site soils and/or sediments (i.e., the residual contamination is fixed).

7.8 Equipment Selection

Special Work Permits (SWPs) and Work Plans (WPs) were written and approved for the field work required during the conduct of this final status survey. The SWPs and/or WPs for this project specified the type of instrumentation to be utilized in performing the site surveys. The instrumentation utilized by site personnel is discussed below.

7.8.1 Equipment and Instrumentation

The instrumentation utilized to generate the characterization and final status survey data is calibrated and maintained in accordance with the Radiation Protection Program procedures. These procedures utilize the guidance contained in ANSI N323-1978, "Radiation Protection Instrumentation Test and Calibration"[23]. Specific requirements for instrumentation include traceability to NIST standards, field checks for operability, background radioactivity checks, operation of instruments within established environmental bounds (i.e. temperature and pressure), training of individuals, scheduled performance checks, calibration with isotopes with energies similar to those to be measured, quality assurance tests, data review, and recordkeeping.

Portable survey instruments utilized during the survey (micro-R survey meters, α/β survey meters, scalers/ratemeters, etc.), are calibrated on a quarterly basis. All instrumentation is calibrated with NIST traceable standards. Where applicable, activities of sources utilized for calibration are also corrected for decay. In addition to the quarterly calibration requirements, source checks are performed on a daily basis for all instruments being utilized for characterization and final status surveys. A calibrated electronic pulse generator is utilized for instrument scale linearity checks. All calibration and source check records are

completed, reviewed, signed off and retained in accordance with Cimarron Quality Assurance Program requirements.

An SWP was written and approved prior to commencement of field work covered under this Final Status Survey Plan. The SWP specified the type of instrumentation to be utilized in performing the site surveys. The instrumentation utilized by site personnel is discussed below.

7.8.1.1 Micro-R Survey Meter

The Micro-R meter utilizes a 1" x 1" NaI/Tl crystal gamma detector and measures exposure rates between 0 and 5,000 μR/h. Background readings are obtained daily at a defined location prior to placing each instrument into service. This instrument was utilized, in general, for determination of exposure rates at both systematic and random locations, and at locations of elevated radiation as identified by gross beta-gamma scans.

Quarterly comparisons and/or confirmatory measurements are obtained routinely to provide information concerning any measurement bias. These comparisons or confirmatory measurements are made using a pressurized ion chamber.

7.8.1.2 Soil Counter (Gamma Spectroscopy)

The Cimarron Soil Counter consists of a 4" x 4" x 16" sodium iodide crystal housed in a shielded chamber which is computer linked to a multi-channel analyzer (MCA). The soil counter is programmed to determine the total uranium concentration present in the soil sample by calculating the U-234 concentration present from the U-235 concentration which is measured in the soil. These two isotopic values are then summed with the measured U-238 value to determine the concentration of total U. Calibration of this counting system is performed annually and is traceable to NIST standards through contractor laboratory evaluations of the on-site standards.

ORISE has been used by the NRC for verification of a majority of the decommissioning work completed to date at the Cimarron site. ORISE has conducted an evaluation of the Cimarron Soil Counting system's ability to accurately measure total uranium concentrations in soil samples. This was done by comparing ORISE sample analysis results obtained by alpha pulse height analysis and gamma spectroscopy with the results obtained from the use of the Cimarron Soil Counter. ORISE and Cimarron analysis results compared favorably at levels above background as demonstrated by the most recent confirmatory analysis performed for the On-Site Disposal Cell, Pit #3 (NRC cover letter dated July 31, 1997)[24]. NRC inspection Report #70-925/97-02, which accompanied this letter, states that "no significant bias or statistical errors between

the license's soil results and the NRC's results were identified". Additionally, the confirmatory analysis performed on select soil samples collected during ORISE's site visit to investigate the South U-Yard[21], and DAP-3 stockpile[25] verified previously that Cimarron's on-site counter results are statistically identical to ORISE's results.

Established quality assurance measures for the soil counter include Cesium-137 centroid checks, Chi-square tests, background determinations, and the counting of soil standards. All of these quality assurance controls are recorded on control charts and are trended on a continuing basis.

Standards used for calibration and quality assurance checks for the soil counter have been analyzed by outside laboratories and are NIST traceable through these analyses. Comparisons have been made between the standards as counted using the soil counter and two off-site independent laboratories. The assigned values for the standards are the average of the results obtained from the off-site independent laboratories, when the standards were analyzed by more than one laboratory. The standards used at Cimarron range in concentration from 4.5 pCi/g total uranium to 292 pCi/g total uranium.

Cimarron personnel determine uranium and thorium activities based upon the evaluation of net counts from the soil counter. Activities are calculated through the use of efficiency and correction factors obtained using appropriate standards. Soil concentrations are calculated by dividing the net activity by the soil mass. Soil masses are determined on a laboratory scale which is checked on a daily basis (when in use) utilizing NIST traceable standards. Corrections for soil moisture content are also made as necessary.

7.9 Procedures/Plans

As discussed in Section 7.8, SWPs and WPs were written and approved for the field work required for this final status survey. These SWPs and WP's are an integral part of this site's radiation protection and quality assurance programs. Project organization and responsibilities, which are a part of the site's quality assurance program, are discussed in this section.

7.9.1 Organization

The final status survey of concrete rubble in Sub-Area "F" was performed by a survey team consisting of qualified personnel from the Cimarron site. The final status survey team operated under the general direction of a Decommissioning Supervisor who served as the Project manager and reported directly to the Site Manager of the Cimarron Facility.

The selection of field measurement equipment and sample collection techniques was performed under the direction of the RSO/Health Physics Supervisor who also reports to the Cimarron Site Manager. Actual field measurements and sample collection were performed under the direction of the Decommissioning Supervisor. The Decommissioning Supervisor was responsible for developing the SWP and WP for this sub-Area with input from the RSO/Health Physics Supervisor and the Cimarron Site Manager. The SWP and WP were reviewed and approved by the Cimarron Site Manager and the RSO/Health Physics Supervisor.

7.9.2 Training

Cimarron Corporation provides continuing training to Cimarron personnel and any other personnel (i.e., contractors, visitors, etc.) who are allowed access to the site. All members of the final status survey team attended an in-house training session on the SWP and WP for the work performed under the final status survey plan. All survey procedures and quality assurance requirements were reviewed during this training session.

7.9.3 Radiation Protection Program

Cimarron Corporation maintains a radiation protection program which meets and/or exceeds all of the applicable regulatory requirements associated with activities conducted under Special Nuclear Materials License SNM-928. The Cimarron Radiation Protection Program currently in place for all decommissioning activities is administered through the use of the following documents:

- Cimarron Quality Assurance Plan and Procedures
- Cimarron Radiation Protection Procedures
- Cimarron Site Health and Safety Plan
- Cimarron Emergency Plan

It is the policy of Cimarron Corporation to perform all work in strict compliance with applicable regulatory and internal requirements. The goal of the Cimarron decommissioning effort is to conduct all operations at a level of excellence which exceeds regulatory requirements. Cimarron staff will continue to exercise appropriate radiation protection precautions throughout the remaining decommissioning work and final survey process.

Independent Kerr-McGee Corporate audits for regulatory and internal requirements are conducted on a periodic basis and include the review of the Cimarron Radiation Protection Program and associated programs. Assessments of program effectiveness also are performed periodically by the Cimarron

RSO/Health Physics Supervisor. Additionally, the Cimarron Radiation Protection Program is inspected for compliance with applicable rules and regulations by NRC Region IV and NRC Headquarters staff.

7.9.4 Cimarron Quality Assurance Program (QAP)

The Cimarron Corporation QAP is an integral part of the Cimarron Radiation Protection Program. A principal component of the QAP is the confirmation of the quality of project work performed during decommissioning by assuring that all tasks are performed in a quality manner by qualified personnel. The Program ensures that samples are collected, controlled, and analyzed in accordance with applicable quality controls to provide adequate confidence that the resulting data accuracy and validity are verifiable. Such quality controls allow for the independent verification of analysis results by a third party review.

The Cimarron QAP is implemented and maintained in accordance with written policies, procedures, and instructions. This Program is administered under the direction of the Quality Assurance Manager. Periodic audits and reviews are conducted to ensure that all aspects of the Program are addressed.

Written procedures, designated as SWPs and WP's, are prepared, reviewed and approved for activities involved in carrying out the decommissioning process. The Sub-Area "F" concrete rubble survey SWP and WP were written in accordance with the Cimarron QAP. These documents designated the type of surveys to be performed, samples to be collected, frequency of sample collection, number of samples to be split with an off-site independent laboratory, and the type of field instrumentation required for the tasks required.

The facility performs its own radiological soil analysis in accordance with written procedures and QA/QC protocols. Field data are gathered and maintained in logs for all samples in accordance with the Cimarron QAP. Necessary data are transferred to the on-site laboratory sample log when the sample is brought to the on-site laboratory for analysis. The sample logs provide a record of sample collection and transport (chain of custody) and are incorporated into the facility quality assurance records.

In addition, off-site independent radiological analysis of split samples (samples are first counted on-site and then sent to an off-site independent laboratory) is an integral part of the Cimarron QAP. Samples sent to an off-site independent laboratory for analysis are accompanied by a chain of custody form in accordance with the Cimarron QAP. These forms provide documentation for all aspects of sample control and are maintained by the Quality Assurance Manager as permanent records.

Sample and survey data are reviewed by the Health Physics Department for accuracy and consistency and are compared to the guideline values. Reviews are performed on a regular basis. When identified, corrections to recognized deficiencies are performed.

Planned and periodic audits of Cimarron's Quality Assurance Program are performed by individuals who do not have direct responsibilities for the areas being audited. Audit results are documented for review by management.

8.0 Survey Findings

Final Status Survey data was generated for the concrete rubble located in Sub-Area "F" in order to demonstrate that this concrete rubble could be unconditionally released. The survey findings, including the methodology employed to evaluate the data, are described in this section.

8.1 Thermoluminescent Dosimeter (TLD) Exposure Rate Data

Thermoluminescent dosimeters were placed at three locations (#AM015, #AM016, and #AM017) near or above the concrete rubble. TLD data for 1996 and 1997 are provided in Tables 8.1 and 8.2, along with data for TLD location #AM014, which is located approximately one half mile south of the facility near the junction of Highways #33 and #74. Location #AM014 represents background. All TLDs were placed at a height of approximately one meter above the ground or concrete surface, and were oriented to face the area to be monitored. Drawing No. 98_TLD depicts these TLD locations.

Table 8.1
TLD Exposure Rate Measurements-1996

TLD#	Description	1Q96 µR/h	2Q96 µR/h	3Q96 µR/h	4Q96 µR/h	96 Ave. µR/h
AM014	Junction of Highways 33 & 74 (Background)	7.6	8.3	5.6	7.0	7.1
AM015	Res. #2, SE of Rubble	8.2	7.0	5.7	6.5	6.9
AM016	Res. #2, Middle of Rubble	9.3	10.2	5.3	6.7	7.9
AM017	Res. #2, Below Rubble	8.5	7.6	5.0	5.7	6.7

Table 8.2
TLD Exposure Rate Measurements-1997

TLD#	Description	1Q97 μR/h	2Q97 μR/h	3Q97 μR/h	97 Ave.* μR/h
AM014	Junction of Highways 33 & 74 (Background)	9.2	6.9	8.8	8.3
AM015	Res. #2, SE of Rubble	6.3	5.9	5.7	6.0
AM016	Res. #2, Middle of Rubble	6.7	6.0	6.4	6.4
AM017	Res. #2, Below Rubble	4.6	4.6	6.1	5.1

*Only the first three quarters of 1997 data were available at the time this report was published.

During 1996, the exposure rate near the concrete averaged 7.2 μR/h at the three indicator locations (#AM015, #AM016, and #AM017), and averaged 7.1 μR/h at the background location (#AM014). The exposure rates during the first three quarters of 1997 were similar, averaging 5.8 μR/h at the three indicator locations, and 8.3 μR/h at the background location. Data for the fourth quarter of 1997 have not yet been received from the contractor and are therefore not available.

The TLD exposure data does not indicate any elevated exposures occurring as a result of the elevated concrete surface contamination. This would be expected considering the low activity levels present in the concrete, and the attenuating characteristics of the concrete for any low energy gammas present due to the low residual uranium. In addition, the concrete provides shielding for naturally occurring gamma emitters within surface soils, and thus exposure rates are expected to be lower over concrete than in other locations. This observation is supported by the survey data presented in this report.

The TLD data also supports the measurements obtained with micro-R meters. The TLD data indicates that the guideline value of 10 micro-R above background is met at each TLD location during 1996 and 1997 to date.

8.2 Soil/Sediment Samples

In accordance with the Work Plan, one soil/sediment surface sample (6" depth) was collected from area upgradient of the main body of concrete rubble, ten surface samples from areas within and beneath the concrete, and two surface samples from areas downgradient of the concrete. The sampling locations and sample results are summarized in Table 8.3. The locations where each soil/sediment sample was collected are shown in Drawing No. 98FCRSS (Appendix I). Sample results do not indicate any samples above the BTP Option #1 guideline concentration of 30 pCi/g, above background. Concentrations of total uranium ranged from 2.9 to 8.6 pCi/g, while total thorium ranged from 0.2 to 1.0 pCi/g. The concentrations in these soil and sediment samples are similar to those found in unaffected areas, and do not indicate that there is any significant contribution occurring as a result of the concrete in the immediate vicinity.

Table 8.3
Soil/Sediment Samples Collected Around Concrete

Sample Number	Grid Location	Total U* (pCi/g)	Total Th* (pCi/g)
FA-535 (upgradient)	1451E-820N	4.2	1.2
FA-536	1419E-839N	2.9	0.9
FA-537	1419E-845N	5.9	0.8
FA-538	1420E-848N	8.6	0.2
FA-539	1412E-855N	7.0	0.8
FA-540	1404E-860N	4.2	0.7
FA-541	1341E-865N	4.6	0.7
FA-543 (downgradient)	1358E-870N	3.9	0.3
FA-544	1387E-870N	4.5	0.7
FA-545	1412E-870N	5.3	1.0
FA-546 (downgradient)	1365E-873N	4.3	0.7
FA-547	1378E-875N	4.4	0.6
FA-548	1370E-878N	3.6	0.6

*Reported measurements include the contribution from natural background.

8.3 Surface Water Samples

Surface water samples were collected upstream and downstream of the concrete on September 30, 1997. Sample results are summarized below in Table 8.4.

The upstream and downstream surface water samples show low levels of naturally occurring concentrations of uranium and thorium. There was no significant difference between the upstream and downstream sample. In addition, the sample data do not indicate any significant differences from the historical data reported for environmental

monitoring location #1205 (See Section 6.4). This indicates that the concrete is not contributing to the concentrations of these naturally occurring contaminants.

Table 8.4
Laboratory Data for Surface Water Samples
(Results in pCi/l)

Location	Th-228	Th-232	U-234	U-235	U-238
Upstream (FA-WAT-315)	0.4±0.3	<0.4	0.5±0.3	<0.1	<0.3
Downstream (FA-WAT-316)	0.4±0.4	<0.5	0.8±0.4	<0.2	0.3±0.2

8.4 Micro-R Measurements

A tabular summary of all micro-R meter surveys for the random grid samples and for the all sampled grids is provided in Tables 1 and 2 of Appendix II. Drawing No. 98FCRER (Appendix I) also presents the average exposure rate measurements for each 5m x 5m grid area for which measurements were collected. For the random sample, the average exposure rate over any 25 m^2 grid area ranged from 6 μR/h to 10 μR/h at one meter from the surface. The overall average exposure rate for the random sample was 7 μR/h at one meter from the surface. At one meter above the surface, the maximum exposure rate was 10 μR/h, including background. This maximum was measured at the location that had the highest exposure rate at the surface, which was 25 μR/h, including background (grid # 52). Assuming a background of 7 μR/h, the net annual exposure rate at this surface location would be 18 μR/h. This location was evaluated to determine any significance with respect to exposure of the general public. Under normal circumstances, it is unlikely that any additional exposure would occur to members of the public as this piece of concrete is within a drainage area, and is on land owned by Cimarron Corporation. The possible exposure scenarios evaluated included hunting the land or an intruder inadvertently remaining in the area for a period of time. Assuming ten hours per year exposure, the hypothetical individual could receive an annual dose of 180 μrem to the portion of the skin of the whole body or to any organs situated directly in contact with the concrete with high residual activity. A more likely scenario is from a person standing in the area for a period of several hours per year. The net annual dose rate above background from this hypothetical activity would be approximately 9 μrem/y to the whole body, based upon the net measured exposure rate at a height of one meter. Both of the above dose scenarios are unlikely, in that the concrete rubble is not in an area where it would be desirable to spend any amount of time. In comparison to the exposure that an individual receives from natural background radiation (≈300 millirem/y), the calculated hypothetical doses of 180 μrem/y and 9 μrem/y are insignificant.

The data summary for exposure rate measurements of all sample grids (i.e., random sample plus supplemental grid area data) is also included in Table 2 of Appendix II. This data is similar to the random grid sample data described above, and also indicates that there are no significant exposure risks from leaving the concrete in place.

The data summarized in Tables 1 and 2 of Appendix II, and presented in Drawing No. 98FCRER, include the exposure due to natural background radiation, which has been previously determined to range from 7-10 μR/h at the Cimarron site. Therefore, the estimated dose at one meter is essentially equal to that which would be received due to natural background.

8.5 Gross Alpha and Gross Beta-Gamma Surface Activity Data

Gross beta-gamma scans were performed over the entire surface of selected random grids containing concrete rubble to determine the nature and extent of the activity. In addition, the random sample data were supplemented with additional measurements to ensure that the random sample was representative. Gross alpha and gross beta-gamma surface activity data are summarized in Tables 3 through 6 in Appendix II. Drawings No. 98FCRA and 98FCRB (Appendix I) also present the average gross alpha and gross beta-gamma surface activity data for each 5m x 5m grid area. The data tables present the measured activity for each representative area and hot spots within the sampled 5m x 5m grids. Where applicable, background subtraction was performed on the gross beta-gamma hot spot data to obtain the expected increase that is due to residual activity. The average and maximum activity (dpm per 100 cm^2) is calculated for each of the 5m x 5m grids. The average volumetric concentration over the 5m x 5m grid area was then calculated using the relationship between gross beta-gamma surface activity and volumetric concentration that is presented in Section 8.6.2.

The volumetric concentration calculation assumes that the average gross beta-gamma surface activity is representative of the grid area as a whole, and that there is equal probability of measurement of the residual activity on the most elevated side of the concrete as there is for measurement on the least elevated side. This assumption is reasonable based upon the random manner in which the concrete was placed, and the random manner in which the sampling of the grid areas was performed. In order to account for the probability for residual activity to exist on both the top and bottom sides of each concrete slab, the average thickness for the concrete was divided by two. Therefore, since the average thickness of the concrete was found to be one foot, the volumetric concentration was calculated over a thickness of six inches.

8.5.1 Gross Beta-Gamma Data

Overall, the random grid sample data contained more elevated locations per grid and was found to be more limiting in that the data indicated higher overall concentrations than the data which included both the random sample and the

supplemental survey data. The maximum gross beta-gamma surface activity (concrete background subtracted, and averaged over $1m^2$) was found within grid #43 (hot spot #1). This hot spot location measured 26,075 dpm/100 cm^2. The highest average gross beta-gamma surface activity over any 5m x 5m grid was found in grid #51, which averaged 4,867 dpm/100 cm^2, with background subtracted. Using the conversion to volumetric concentration, this equates to 7.4 pCi/g average total uranium concentration over the $25m^2$ area. This concentration is well below the BTP Option #1 guideline value.

The overall average volumetric concentration calculated for the random sample was 2.9 pCi/g. For all sampled grids (i.e., random sample and supplemental samples), the average volumetric concentration was found to be 1.8 pCi/g. Although above typical background levels for concrete, these concentrations are similar to those found in nature and are well below the BTP Option #1 criteria. Therefore, the health and safety significance of leaving the concrete in place is similar to the health and safety considerations for natural soils. The potential uses of the concrete are limited by its portability and by the difficulty that would be experienced through attempts to remove it from the drainage areas. Therefore, it is anticipated that the any exposures to the concrete would be from casual contact or from its gradual disintegration over time due to environmental forces.

8.5.2 Gross Alpha Data

The gross alpha survey data are presented in Tables 5 and 6 in Appendix II and are also summarized in Drawing No. 98FCRA (Appendix I). Data are summarized for the random sample as well as for all sampled grids. Data generally indicate that the concrete rubble would meet the current gross alpha guideline criteria of 5,000 dpm/100 cm^2 (average) and 15,000 dpm/100 cm^2 (maximum) for unconditional release. As previously discussed in Section 1.0, the concrete was placed into Sub-Area "F" after gross alpha surveys were performed. The criteria in effect at the time the concrete was placed into Sub-Area "F" varied from the criteria currently in effect (as discussed above) up to 25,000 dpm/100 cm^2, which was allowable under the NRC criteria which was in effect at the time.

Review of the data indicate that two of the sampled grids contained hot spots exceeding 5,000 (average) or 15,000 (maximum) dpm/100 cm^2. Grid #52 contained a hot spot with 9,000 dpm/100 cm^2 (average) and 16,000 dpm/100 cm^2 (maximum). In addition, Grid #56 contained a hot spot with 15,600 dpm/100 cm^2 (maximum).

The criteria proposed for release of the concrete is based upon volumetric concentration. The concrete rubble does not have any smearable contamination and the activity would have to be removed through mechanical or physical forces. While it is probable that environmental forces will eventually act to remove the radioactivity,

normal environmental dispersion will result in insignificant quantities available for ingestion or inhalation.

8.6 Calculations

This section describes calculations and evaluations that were required in order to evaluate the concrete rubble and perform comparisons with the proposed release criteria. The calculations included determinations of the average thickness of the concrete, which was used to determine the appropriate volume of concrete for averaging the residual activity. The calculations performed to determine the relationship between gross beta-gamma surface activity and volumetric concentration of total uranium are also described in this section. Finally, source term was estimated based upon the random sample data. The source term allows for evaluation of the acceptability of the proposed action, which is to leave the concrete in place. The source term can be utilized for input into computer models or for use in comparing the residual activity with that present in the natural environment. A RESRAD computer model was run to calculate hypothetical dose to individuals over a period of 1000 years.

8.6.1 Calculation of Average Concrete Thickness

The volume estimate for the concrete was previously presented in the Decommissioning Plan. The data was evaluated and a weighted average thickness for the concrete was calculated. The weighted average thickness accounts for the presence of different volumes of concrete rubble that also have different thickness. The weighted average is calculated as follows:

$$\text{Weighted Average} = \sum[(\text{thickness}) \times (\text{volume})]/\sum[\text{volume}];$$

The above formula can be explained as follows. For each area of the facility, the concrete thickness in the area is multiplied by the volume of the concrete that came from the area. These individual volume thickness calculations are summed and divided by the overall volume of the concrete (i.e., all concrete that was placed into Sub-Area "F") to obtain the weighted average.

Using the above calculation, the weighted average thickness of the concrete in Sub-Area "F" was estimated to be 1 foot. The data tables showing the dimensions of the concrete present in Sub-Area "F" are presented in Appendix III.

8.6.2 Volumetric Concentration Conversion Factor

A special study was performed as described in Section 7.6 to determine a relationship between gross beta-gamma surface activity on the concrete (measured in dpm/100cm^2) and the volumetric concentration (measured in pCi/g) of total uranium. The solution to this problem is confounded due to the natural presence of beta and gamma emitters in concrete, including uranium. In addition, the low energy of the beta emitters

associated with enriched uranium and the variability in depth of the contaminated layer hinder the determination of a single conversion factor for this purpose. However, the data presented for the two slabs studied indicate a reasonable agreement between the data.

For this study, two slabs were selected. Slab #1 had an initial average gross beta-gamma surface activity of 4496 dpm/100cm^2 (concrete background subtracted), while slab #2 had an initial average gross beta-gamma surface activity of 17,697 dpm/100cm^2 (concrete background subtracted). Measurements were performed on each slab as described in Section 7.6, followed by scabbling of another layer (average depth approximately 1/8 inch), resurvey, and additional scabbling until the majority of the contamination was removed. The scabble dust and particles were collected and analyzed for total uranium content using either the onsite soil counter or using an off-site independent laboratory. Survey and laboratory data for the two slabs is presented in Table 8.5.

The special study data revealed that the maximum depth of the contamination was approximately 3/8 inch. The contamination was found to be highest, as expected, at the layer closest to the surface, and decreased substantially as additional layers of concrete were scabbled and removed.

The concentration of total uranium was assumed to be at background (i.e., 1.5 pCi/g) for all remaining concrete when the gross beta-gamma measurements indicated that the residual activity had been essentially removed. Complete removal of the residual activity was achieved after two scabbling operations on Slab #1, and after three scabbling operations on Slab #2. It was determined that each scabbling operation removed approximately 1/8 inch from the surface. The volumetric concentration conversion factor was determined using the average thickness of six inches as follows:

<u>Slab #1</u>

Volumetric average conversion factor (dpm/100 cm^2 gross beta-gamma per pCi/g total U) =

(17,697 dpm/100 cm^2) ÷ {[814.7 pCi/g + 102.7 pCi/g + (1.5 pCi/g x 46)] ÷ 48} =

<u>861.2 dpm/100 cm^2 gross beta-gamma per pCi/g total U.</u>

TABLE 8.5

Survey Data for Concrete Rubble Slabs Used to Determine the Volumetric Concentration Conversion Factor

	Gross Alpha dpm/100cm^2 (ave)	Gross Alpha dpm/100cm^2 (max)	Gross Beta dpm/100cm^2 (ave)	Gross Beta Dpm/100cm^2 (max)	Slab Surface µR/h	Slab 1m µR/h	Hot Surface µR/h	Spot 1m µR/h	Total U Conc. (pCi/g)
Slab #1									
Initial Measurement	1251	4800	17,697	44,540	9	7	11	7	814.7
After 1st Scabbling	157	480	2411	8330	6	6	6	6	102.7
After 2nd Scabbling	24	160	0	1310	6	6	6	6	
Slab #2									
Initial Measurement	329	1280	4496	24,390	6	6	9	8	313
After 1st Scabbling	115	480	935	13,370	5	6	8	6	74.9
After 2nd Scabbling	116	400	0	4,950	6	6	6	6	17.7
After 3rd Scabbling	<350	<350	0	230	9	9	9	9	

Notes:
1) Concrete gross beta-gamma background (800 dpm/100cm^2) subtracted from beta-gamma measurements.
2) No background subtraction performed for all other measurements.
3) Measurements less than 0 after background subtraction were recorded as 0.
4) Total U concentration assumed as 1.5 pCi/g when all residual gross beta-gamma activity was determined to be removed by scabbling.

Slab #2

Volumetric average conversion factor (dpm/100 cm^2 gross beta-gamma per pCi/g total U) =

(4496 dpm/100 cm^2) ÷ {[313 pCi/g + 74.9 pCi/g + 17.7 pCi/g + (1.5 pCi/g x 45)] ÷ 48} =

456.2 dpm/100 cm^2 gross beta-gamma per pCi/g total U.

In the above calculations, each scabbled layer is assigned a thickness of 1/8 inch, which corresponds to the measured concentration of the concrete. It follows that there are 48, 1/8 inch layers in a six inch slab of concrete. Each background layer is assumed to have a total uranium concentration of 1.5 pCi/g, as discussed in Section 6.2.1. The numerator in the above equations is the measured gross beta-gamma surface activity on the top layer. This data is readily available and was obtained during the surveys of the concrete rubble. The two slabs studied in this special project indicate that the conversion factor is in the range of 456 to 861 dpm/100 cm^2 gross beta-gamma per pCi/g total U. The two samples resulted in calculated conversion factors that were within a factor of two, which is good agreement considering the numerous areas of uncertainty.

The average of the two measurements, which is 661 dpm/100 cm^2 gross beta-gamma per pCi/g total U, was utilized to estimate the average volumetric concentration of residual activity present in the concrete. For comparison, the volumetric conversion factor was also calculated for an assumed concrete thickness of three inches. This conversion factor was also used to calculate average total uranium concentrations. The data tables for gross beta-gamma surface activity in Appendix II (Tables 3 and 4) also present the average volumetric concentrations for each 5m x 5m grid area.

8.6.3 Volumetric Concentration Calculations

Calculation of the estimated volumetric concentration (in pCi/g total U) was performed by multiplying the conversion factor described in Section 8.6.2 times the measured gross beta-gamma surface contamination measurement (with background subtracted). These calculations, which are summarized in Appendix II (Tables 5 and 6), resulted in average total uranium concentrations (after background subtraction) ranging from – 0.8 pCi/g to 7.4 pCi/g. The negative results indicate that the 5m x 5m grid average was less than 1.5 pCi/g. Subtraction of the 800 dpm/100cm^2 concrete background thus resulted in a value that was less than average. Assuming a normal distribution of a background distribution, one half of the samples collected would be expected to be less than background.

The maximum 5m x 5m grid average total uranium concentration was 7.4 pCi/g (grid #51). This concentration is less than 25% of the guideline value of 30 pCi/g. The average total uranium concentration for the random sample was 2.9 pCi/g, which is less than 10% of the guideline value for enriched uranium (30 pCi/g). For the random

sample and the supplemental grid areas (66 grid areas), the total uranium average concentration was 1.8 pCi/g, which is equal to 6 percent of the guideline value.

8.6.4 Source Term Calculation

The average volumetric concentration for the random sample was calculated to be 2.9 pCi/g. Since the volume of concrete is known, an estimate of the total activity of uranium present in the concrete can be calculated as follows:

Total activity = (2.9 pCi/g-concrete) x (1.8 g-concrete/cc) x 31,985 ft^3 concrete x 28,320cc/ft^3

= 4.7 E+09 pCi = **4.7 E-03 Ci Total Uranium.**

8.6.5 Pathway Analysis

The RESRAD computer code was used to evaluate the potential dose due to leaving the concrete in place. The RESRAD code considers direct radiation, inhalation of resuspended radioactivity, ingestion of groundwater and foodstuffs grown in contaminated soils, or in soils irrigated with contaminated surface or ground water, and all other credible pathways. The RESRAD model generally will predict a more conservative dose (i.e., a higher dose) than that which could potentially be received, as it generally utilizes conservative assumptions and includes scenarios for use of the land area that are generally not consistent with the expected uses for concrete rubble.

The input parameters for RESRAD include those defined in NRC's Policy and Guidance Directive (PG) 8-08, "Scenarios for Assessing Potential Doses Associated with Residual Activity"[26]. The uranium isotopic ratios were chosen to be the same as those used by the NRC for the "Environmental Assessment Associated with the BTP Option #2 Onsite Disposal Cell at Cimarron"[27], which were U-234 (79%), U-235 (1.7%), and U-238 (20%). The selected density for the concrete was 1.8 g/cc. The calculated area of the contaminated zone is 2970 m^2 [31,985 ft^3 x $(0.3048m/ft)^3$ ÷ 0.3048m], while the calculated thickness is 0.3048m.

The RESRAD calculated maximum dose rate will occur at 900 years and result in a maximum hypothetical annual dose to the resident of approximately 1 millirem per year. A printout of the parameters used and results of the RESRAD calculation are provided in Appendix IV.

9.0 Summary

A Final Status Survey was performed on the concrete in Sub-Area "F". The survey incorporated NRC guidance and suggestions for volumetric concentration averaging. This Report presents the results of the Final Status Survey. The survey data were evaluated and doses from leaving the concrete in place were calculated. The evaluations presented in this report indicate that the concrete should be left in place for the following reasons:

- **RESRAD dose evaluation indicates that the projected maximum dose would occur after 900 years, and that this dose would be approximately 1 millirem per year.**

- **Random sample data indicate that the average total uranium concentration for the concrete rubble is 2.9 pCi/g. The overall average total uranium concentration in the concrete is less than 10 percent of the BTP Option #1 guideline value (30 pCi/g). The average total uranium concentration, based upon all sample data, was calculated to be 1.8 pCi/g.**

- **The maximum concentration averaged over any 5m x 5m grid area was found to be 7.4 pCi/g., which is less than 25% of the BTP Option #1 guideline value.**

- **The overall average exposure rate at one meter from the surface was 7 μR/h, which is equivalent to natural background. The maximum exposure rate at one meter from the surface was 10 μR/h, which is within the range of natural background.**

- **The calculated total uranium source term is 4.7 millicuries within the total volume of concrete rubble estimated at 31,985 ft^3.**

- **No elevated measurements observed for exposure rate as measured by thermoluminescent dosimeters placed in the field adjacent to the concrete rubble.**

- **Surface water samples collected upstream and downstream did not indicate any contribution of radioactivity from the concrete rubble. Soil/sediment samples collected upstream, within, and downstream of the concrete rubble reflect radioactivity levels which are characteristic of samples collected in unaffected areas of the facility.**

- **The concrete continues to serve the useful purpose of on-site erosion control.**

The calculations presented in this report utilized conservative assumptions. Therefore, it is unlikely that even the low dose which was calculated could be received by any member of the public. Based upon the evaluations in this report, Cimarron Corporation requests authorization from the NRC for unconditional release of the concrete.