environmental
properties management, LLC

May 7, 2019

Mr. Ken Kalman
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Mr. Paul Davis
Oklahoma Department of Environmental Quality
707 North Robinson
Oklahoma City, OK 73101

Mr. Robert Evans
U.S. Nuclear Regulatory Commission
1600 East Lamar Blvd; Suite 400
Arlington, TX 76011-4511

Re: Docket No. 70-925; License No. SNM-928
Response to February 28, 2019 Request for Supplemental Information

Dear Sirs:

Solely as Trustee of the Cimarron Environmental Response Trust (the CERT), Environmental Properties Management, Inc. (EPM) provides responses to Nuclear Regulatory Commission (NRC) requests for supplemental information in your letter dated February 28, 2019. It is our understanding that this additional information is needed to enable the NRC to complete a detailed technical review of the Cimarron *Facility Decommissioning Plan – Rev 1* (the DP).

Attachment 1 to this letter provides each NRC request for information, followed by EPM's response. Information provided includes responses to each information request and indicates where revisions to specific sections of the DP or RPP will be made. Attachment 2 presents proposed revisions to the DP that address groundwater treatment or remediation issues. Attachment 3 provides proposed revisions to the DP that address technical radiological protection and license compliance issues. Attachment 4 provides proposed revisions to the RPP. Attachments 2 and 3 are separate, because those responses were generated by different entities on behalf of EPM.

The responses to requests for information also indicate where additional information is provided. Information such as calculations which support statements in the DP or RPP, but which should not be incorporated into the DP or the RPP, provide information supporting statements made in the responses. This information is provided in the following enclosures:

- Enclosure A – Groundwater Model Input Files
- Enclosure B – May 3, 2019 Letter to NRC and DEQ Regarding Tc-99 in Influent, Effluent, and Waste

- Enclosure C – Radiation Protection Procedures Related to Radiological Instrumentation
- Enclosure D – Procedure RP-10, "ALARA Program"
- Enclosure E – Quality Assurance Program Plan, Revision 4
- Enclosure F – Dosimetry Data – 2013 Through 2018

The NRC requested the native files for the groundwater flow models. Because these native files need to be submitted in electronic format, the decision was made to provide all Attachments and Enclosures in electronic format on a Digital Video Disc (DVD). Only Attachment 1, the responses to the NRC's requests for supplemental information, is attached to this letter in hard copy format.

Should you have any questions or desire clarification of this response to NRC's request, please contact me at jlux@envpm.com or 405-642-5152. Thank you.

Sincerely,

Jeff Lux

Jeff Lux, P.E.
Project Manager

cc: NRC Public Document Room (electronic copy only)

Attachment 1:	EPM Responses to February 28, 2019 Request for Supplemental Information
Attachment 2:	Proposed Revisions to Facility Decommissioning Plan – Rev 1 Related to Groundwater Treatment or Remediation Issues
Attachment 3:	Proposed Revisions to Facility Decommissioning Plan – Rev 1 Related to Technical Radiological Protection Issues
Attachment 4:	Proposed Revisions to the Cimarron Radiation Protection Plan, Rev. 4

Enclosure A:	Groundwater Flow Model Input Files
Enclosure B:	May 3, 2019 Letter to NRC and DEQ Regarding Tc-99 in Influent, Effluent, and Waste
Enclosure C:	Radiation Protection Procedures Related to Radiological Instrumentation
Enclosure D:	Procedure RP-10, "ALARA Program"
Enclosure E:	Quality Assurance Program Plan, Revision 4
Enclosure F:	Dosimetry Data – 2013 Through 2018

ATTACHMENT 1

EPM Responses to 2/28/2019 Request for Supplemental Information

SECTION 2.7.6: GROUNDWATER MODELS

The licensee provided a description of the numerical analyses techniques used to characterize the unsaturated and saturated zones. The licensee stated that a previously developed groundwater numerical model has been revised and updated for the Cimarron site. The updated groundwater numerical model was used to support groundwater remedial design in terms of layouts of extraction wells and pumping rate. Consequently, the input parameters to the numerical model may have been revised. An updated discussion of the groundwater flow model needs to be provided in Facility Decommissioning Plan – Rev 1 (DP), including the numerical groundwater input files so that NRC staff can independently verify the modeling results.

> **EPM Response:** Section 2.7.6 of *Facility Decommissioning Plan – Rev 1*[1] (DP) will be revised to include additional details regarding the updates to the numerical groundwater flow models and input parameters. No structural changes have been made to the groundwater flow models since 2016; the 2016 model updates were documented in the *2016 Groundwater Flow Model Update*[2]. Native numerical groundwater model input files are included as Enclosure A. These files were generated using Groundwater Vistas (Version 7). Proposed revisions to the DP are included in Attachment 2.

SECTION 3.5: GROUNDWATER

The licensee referenced reports describing the groundwater assessment, but a summary of the impacted aquifers in separate areas of the site is not explicitly included. Please provide a brief description of the aquifers that need to be remediated.

> **EPM Response:** Section 3.5 of the DP will be revised to include a description of the aquifers targeted for remediation activities, as described in the Section 3 figures. Proposed revisions to the DP are included in Attachment 2.

SECTION 3.5.3: CURRENT EXTENT OF CONTAMINANTS OF CONCERN IN GROUNDWATER

The maximum and average radionuclide activities are shown in Figures 3-1 through 3-4 of the DP. However, a description of the radionuclide activities in each of the aquifers is not adequately described. The magnitude and extent of uranium activities in groundwater, including the maximum and average uranium activities, and the recoverable amount of uranium in the aquifer in the BA-1 and the western area should be addressed in the DP.

> **EPM Response:** The magnitude and extent of representative uranium mass and activity concentrations in groundwater in the Western Area and Burial Area 1 are presented in Figures 3-3 and 3-4 of the DP, respectively. Proposed revisions to Section 3.5.3 of the DP include paragraphs describing the magnitude and extent of uranium activity concentrations in groundwater, as presented in Figures 3-3 and 3-4. The maximum and average uranium concentrations for each aquifer within the Western Area and Burial Area 1 (BA1), and the maximum and average uranium concentrations for each

---

[1] *Facility Decommissioning Plan – Rev 1,* Environmental Properties Management LLC, November 2018

[2] *2016 Groundwater Flow Model Update,* Burns & McDonnell, January 2017

remediation area exceeding the NRC remediation criterion will also be presented in these paragraphs. These maximum and average concentrations will be provided in mass concentration form as activity concentrations have not been calculated for individual monitor wells or remediation areas. During discussions with NRC and DEQ on May 5, 2019, NRC stated that it would not be necessary for EPM to provide uranium activity concentrations in response to this request and that mass concentrations would be acceptable. Proposed revisions include the estimated mass of uranium that will be recovered from each remediation "sub-area" exceeding the NRC remediation criterion, from the time remediation begins until the NRC remediation criterion is achieved. It is understood that the total mass of uranium removed over the course of site remediation will be greater than the mass of uranium recovered from remediation sub-areas exceeding the NRC Criterion because uranium will also be recovered from areas in which the concentration of uranium is below the NRC Criterion, but above the State Criterion. Proposed revisions to the DP are included in Attachment 2.

SECTION 5.6.11: LAND USE

The licensee stated (DP 5.6.11; Radiation Protection Plan (RPP) 6.9) that its annual administrative ALARA goal is 100 mrem TEDE Please describe how this goal is verified as the workplace air sampling program triggers are significantly higher than this value.

EPM Response: Section 6.9 of the Radiation Protection Plan, Rev. 4, (RPP) specifically states that ALARA goals will be set "if individual monitoring is required." Potential exposures from external and internal sources have been evaluated as discussed subsequently in responses to information requested on Section 11 of the DP. Based on these evaluations that consider the groundwater treatment system design and groundwater contamination levels, CERT has determined that individual monitoring is not required. The ALARA Committee has established an ALARA goal of 100 mrem/yr because that is the maximum dose for a member of the public.

Regarding the trigger levels for air sampling, Section 6.1 of the RPP states that air sampling during spent ion exchange resin handling activities will be performed. This is further clarified in Section 10.6, which acknowledges that general area air sampling will be performed throughout the resin unloading and packaging process for at least the first three resin exchanges. Following analysis of the air sample results, the Radiation Safety Officer will determine the need for additional air sampling. Additional air sampling and individual monitoring would be considered if air sampling results indicate a potential for an individual to receive an intake of 100 mrem CEDE in a year. Additional discussion related to the air sampling program and trigger values is provided in responses to information requested in Section 11.2 of the Decommissioning Plan.

SECTION 8.6: IN-PROCESS MONITORING

Monitoring groundwater treatment progress is discussed by the licensee (DP 8.6, 15.3). Please provide an analysis by the licensee demonstrating that discharges are in accordance with the OPDES Discharge Permit and meet the requirement of 10 CFR 20.2001. This would include an analysis based on enriched uranium and the unity rule, taking Tc-99 into account. Please include a discussion of Tc-99 related to discharges into the Cimarron River. Please provide an analysis that demonstrates effluent discharges are in compliance with 10 CFR 20.2001 or identify where this demonstration is in the application.

EPM Response: The primary contributor to the beta activity detected in groundwater at the Cimarron site is believed to be Tc-99, which is present above detection limits only in the Western Area. The quantity and distribution (spatial and temporal) of Tc-99 groundwater data in the Western Area are limited and samples have not been analyzed for Tc-99 since 2012. However, in response to this request for supplemental information, CERT evaluated available Tc-99 concentration data to provide an estimate of the anticipated Tc-99 concentration in the Western Area Treatment Facility (WATF) influent stream. Using the available Tc-99 groundwater results for samples collected between 2003 and 2012, an estimated influent WATF Tc-99 concentration of 466 pCi/L was calculated for the purposes of evaluating compliance with 10 CFR 20.2001 and the potential for Tc-99 to accumulate in WATF ion exchange resin and biodenitrification biomass waste. Detailed descriptions of the historical sources of Tc-99 at the site, the available data set, and the WATF influent concentration calculations are provided in the enclosed letter addressed to NRC and DEQ and dated May 3, 2019. The letter is included as Enclosure B.

The EPA has determined that 900 pCi/L represents the Tc-99 activity concentration that equates to the maximum contaminant level (MCL) of 4 millirem per year for beta emitters. The NRC has determined that a Tc-99 concentration of 3,790 pCi/L equates to the 4 millirem per year MCL. Because the expected WATF influent Tc-99 concentration was expected to comply with the lower of the two drinking water standards described above, Tc-99 was not addressed in the OPDES permit application; consequently, OPDES Discharge Permit OK0100510 (included as Appendix H to the DP) does not stipulate a limit for Tc-99, nor does it require monitoring for Tc-99.

The Oklahoma Water Resources Board (OWRB) has established water quality standards for designated beneficial uses of surface water. The OWRB determines what beneficial uses apply to surface water in the State of Oklahoma. There are no water quality standards for either Tc-99 or gross beta for any of the Cimarron River's designated beneficial uses. However, for a public water supply, the OWRB has established a water quality standard of 50 pCi/L for gross beta. Consequently, the DEQ may be evaluating this value as a relevant and appropriate discharge permit limit for the WATF effluent to Outfall 001. Based on available data there is no evidence of Tc-99 impact to the Cimarron River. Upcoming consultation with DEQ regarding the potential presence of Tc-99 in the WATF effluent will determine if formal notification or an application for a permit modification is needed.

To evaluate compliance with 10 CFR 20.2001, CERT conservatively assumed that no Tc-99 will be removed during treatment of the WATF influent groundwater prior to discharge. The effluent limits for uranium and Tc-99 listed in 10 CFR Part 20 Appendix

B, are 3E-7 and 6E-5 µCi/mL, respectively. These concentration limits are equivalent to 300 and 60,000 pCi/L, respectively. Veolia has asserted that uranium should be non-detectable in WATF effluent, but for this evaluation, CERT conservatively assumed an effluent uranium concentration of 5 pCi/L. Using the average Tc-99 concentration of 466 pCi/L presented above, a unity rule analysis for estimated activity concentrations of uranium and Tc-99 in effluent discharges is as follows:

$$\sum = \left({5\frac{pCi}{L}} \middle/ {300\frac{pCi}{L}} \right) + \left({466\frac{pCi}{L}} \middle/ {60{,}000\frac{pCi}{L}} \right) = \mathbf{0.024}$$

Uranium and Tc-99 concentrations in effluent will be a small fraction of NRC-stipulated effluent limits. The DP will be revised to include this demonstration of compliance with 10 CFR 20.2001. In addition, the analysis of Tc-99 in WATF influent and ion exchange treatment system effluent will be added to Table 8-3b. The analysis of Tc-99 in WATF effluent will be added to Table 8-3c if modification of the OPDES permit requires monitoring of effluent for Tc-99; if needed, this change will be made in accordance with license condition 27(e). Proposed revisions to the DP are included in Attachment 2.

As in the enclosed letter addressed to the NRC and the DEQ and dated May 6, 2019, EPM intends to conduct a comprehensive, synoptic sampling and analysis event to evaluate current Tc-99 concentrations in groundwater at the Cimarron site. EPM also proposes to conduct a treatability test using site-specific groundwater and the ion exchange resin selected for uranium removal to assess the potential for the ion exchange resin to remove Tc-99 from the influent groundwater. The results of the sampling event and treatability study will be used to further assess OPDES discharge compliance and the potential for Tc-99 to accumulate in WATF ion exchange resin and biodenitrification biomass waste. The results are not anticipated to affect conclusions related to the demonstration of compliance with 10 CFR 20.2001 presented above.

SECTION: 11: RADIATION PROTECTION PROGRAM

In Section 11 of the DP, the licensee referred to the RPP as Appendix O. However, Appendix O is titled *Criticality and Uranium Loading Calculations.* The RPP is located in Appendix N. Please explain how this will be corrected and make the correction.

EPM Response: The DP will be revised to correct the reference for the RPP from Appendix O to Appendix N. This revision to the DP is included in Attachment 3.

SECTION 11.1: AIR SAMPLING PROGRAM

1) The licensee stated (DP 11.1; RPP 10.1, 10.6): *"U.S. NRG Regulatory Guide 8.25,* "Air Sampling in the Workplace *provides an acceptable method for meeting certain survey and dose* assessment *requirements of 10 CFR 20. Air samples shall be collected whenever the airborne radioactivity levels are expected to exceed 10 percent of the Derived Air Concentration (DAC) as listed in Appendix B, Table 1 "Occupational" of 10 CFR 20."*

Please provide a description of how airborne radioactivity levels are estimated (e.g., Section 1 of NUREG-1400, and Regulatory Position 1.1 and Table 1 of Regulatory Guide 8.25).

> **EPM Response:** Using the NUREG-1400 methodology, CERT has evaluated the need for air sampling. The maximum amount of U-235 that can be possessed under the license is 1,200 grams. Analyses indicate that the maximum loading of a spent uranium bed is less than 500 grams of U-235 at approximately 1.3% enrichment. Assuming 1.5% enrichment, a total mass of 33 kilograms of uranium would be loaded onto that resin bed. It was further assumed that 12 beds were exchanged per year. As discussed in NUREG-1400, the total activity of uranium handled would exceed 10,000 times the ALI. Following the methodology, the potential intake from handling this amount of uranium was evaluated and determined to be less than 0.1% of the ALI for uranium.
>
> As discussed above, air sampling during spent ion exchange resin handling activities will be performed for at least the first three resin exchanges. The results of the air sampling will be used to verify that the potential intake during a year is less than 0.1% of the ALI for uranium and to determine the need for additional air sampling.
>
> Concentrations of Tc-99 received from processed recycled uranium are detectable (at significantly less than the NRC Criterion of 3,790 pCi/L) at the Site. Even at 3,790 pCi/L, any potential intake would be an extremely small fraction of the ALI and total quantities of groundwater processed at this concentration in a year would not require air sampling or internal monitoring.
>
> Section 10.6 of the RPP will be revised to acknowledge that the evaluation of potential intake was considered in the air monitoring program presented in the RPP. The supporting calculation will be included as Appendix A to the RPP. Proposed revisions to the RPP Table of Contents, page 10-33, and Appendix A are included in Attachment 4.

2) RG 8.25, Table 1, recommends air sampling when the airborne radioactivity levels are expected to exceed 1 percent of the DAC for a worker with an estimated intake less than 10 percent of the applicable annual limit of intake (ALI).

Please provide the technical basis for using 10 percent of the DAC as a trigger for air sampling.

> **EPM Response:** As indicated above, air sampling will be performed throughout the resin unloading and packaging process for at least the first three resin exchanges. Following analysis of the air sample results, the Radiation Safety Officer will determine the need for additional air sampling.
>
> The RPP will be changed to require air sampling when the airborne radioactivity levels are expected to exceed 1% DAC. Changes will be made on pages 10-33, 10-35, and 10-36 of the RPP. Proposed revisions to the RPP are included in Attachment 4.

SECTION 11.2: RESPIRATORY PROTECTION

The respiratory protection triggers (DP 11.2; RPP 14) appear undefined, and potentially too high, as the licensee has not discussed compliance with 10 CFR 20.1201(e) for soluble uranium (if applicable) at the site (see Workplace Air Sampling Program, (2) above). Also, there is no discussion of the details of a potential future respiratory protection program (i.e., consistent with RG 8.15, NUREG/CR-0041, etc.). Please address respiratory protection triggers and a description of a potential respiratory protection program as discussed above.

> **EPM Response:** Evaluations based on the NUREG-1400 methodology related to the need for air sampling do not support the need for a respiratory protection program for radiological purposes. If air sampling results identify unexpected airborne radioactivity levels, Section 10.6.1.5 of the RPP states that action levels will be developed that will include specific action levels for assigning respiratory protection, bioassay analysis, and stopping work.
>
> If a respiratory protection program is required, section14.1 of the RPP incorporates a commitment to 10 CFR 20 Subpart H. Appropriate guidance, such as Regulatory Guide 8.15 and NUREG/CR-0041, would be considered if such a program was necessary. Section 14.2 states that respiratory protection would be used if an individual could be exposed to 40 DAC-hrs in a week or 1 DAC. Airborne radioactive material areas are posted if an individual is likely to receive to 12 DAC-hrs in a week (7 consecutive days starting Sunday) or 0.6% of the ALI in a week.
>
> Section 11.2 of the DP and Section 14 of the RPP will be revised to clarify that the need for a Respiratory Protection Program would be based on a prospective evaluation. If this evaluation indicates the need to post work areas as Airborne Radioactivity Areas, then the program would be implemented based on guidance provided in Regulatory Guide 8.15 and NUREG/CR-0041. Proposed revisions to the DP are included in Attachment 3. Proposed revisions to the RPP are included in Attachment 4.

SECTION 11.3: INTERNAL EXPOSURE DETERMINATION

1) The licensee stated (DP 11.3): "Bioassay sampling will also be performed whenever it is likely that an individual may have received an intake of 10 milligrams of uranium in any one week." Please describe how compliance with the weekly intake of soluble uranium is determined by measurements of airborne radioactive materials.

> **EPM Response:** Based on the evaluation previously discussed using the NUREG-1400 methodology, the potential worker intake for an entire year is less than 0.04 milligrams from inhalation. Drinking groundwater at the maximum concentrations found onsite could result in an intake of 10 milligrams in a week, but consumption of groundwater is not permitted at the site.
>
> The RPP will be changed to include the calculation in Appendix A to the RPP. The proposed revision is provided in Attachment 4.

2) The licensee did not provide (DP 11.3, 11.5; RPP 6. 7, 6.8) a description of how the internal dose to an embryo/fetus will be determined. Please provide a description of how the internal dose to an embryo/fetus will be determined or provide NRC staff with a description of where this information is located in the application.

> **EPM Response:** Current design evaluations have been discussed and internal monitoring for declared pregnant workers is not required. Accordingly, dose to the embryo/fetus from intakes will not be performed. However, if the need to perform internal monitoring for declared pregnant worker is indicated, dose to the embryo/fetus will be determined based on guidance provided in Regulatory Guide 8.36 and ICRP Publication 88.
>
> Section 6.7 of the RPP will be revised to indicate that procedures for determining dose to the embryo/fetus will be developed if internal monitoring for declared pregnant workers is required. These procedures will be based on the guidance discussed. Proposed revisions to the RPP are included in Attachment 4.

3) The licensee stated (DP 11.3): "In addition to the requirements set forth in the RPP, RP procedures include requirements for how worker intakes are determined ... "

Consistent with NUREG-1757, Vol.1, Rev. 2, Section 17.3.1.3, please provide the NRC staff with information regarding how worker intakes are determined including how airborne concentrations are converted to determine intake, etc.

> **EPM Response:** In the absence of a bioassay program, if necessary, intakes to workers would be determined based on representative air sample results. Section 6.6 of the RPP states that procedures will be developed if the need for a bioassay program is identified. These procedures will include provisions for using bioassay results to worker intakes or air monitoring results. Typically, intake from air samples would be calculated by multiplying the activity collected on the air filter by the ratio of the worker's breathing rate (assuming standard man breathing rate for moderate work) to air sampler flow rate.
>
> Proposed revisions to Section 11.3 of the DP, clarifying that procedures do not exist as they will not be required unless a bioassay program is implemented, are included in Attachment 3.

SECTION 11.4: EXTERNAL EXPOSURE DETERMINATION

The licensee stated (DP 11.4): " ... RP procedures describe the type, range, sensitivity, and accuracy of required of individual-monitoring devices. RP procedures also include a description of the action levels for worker's external exposure, and the technical bases and actions to be taken when they are exceeded." Consistent with NUREG-1757, Vol.1, Rev. 2, Section 17.3.1.4, please provide the NRC staff with information regarding the type, range, sensitivity, and accuracy of each individual monitoring device and a description of the action levels for workers' external exposure, and the technical bases and actions to be taken when they are exceeded.

EPM Response: Based on an evaluation of the groundwater processing system at 60% design, workers are not likely to be exposed to external dose requiring the need for personnel monitoring. The RPP will be revised to add Appendix B, containing the supporting calculation (EPM017-CALC-001). Section 11.4 of the DP will be revised to clarify that the procedures referred to would be developed to describe the type range, sensitivity, and accuracy of required individual monitoring devices and that procedures would also include a description of the action levels for worker's external exposure, and the technical bases and action to be taken when they are exceeded.

Proposed revisions to the RPP are included in Attachment 4.

SECTION 11.5: SUMMATION OF INTERNAL AND EXTERNAL EXPOSURE

The licensee stated (DP 11.5; RPP 6.1) that internal monitoring is required for " ... any declared pregnant worker who is likely to receive during the entire pregnancy, a committed effective dose equivalent exceeding 0.1 rem."

10 CFR 20.1208 specifies the maximum dose equivalent to the embryo/fetus. It is not clear how the committed effective dose equivalent to the pregnant worker is related to the dose equivalent to the embryo/fetus. Please provide a description of how workplace monitoring for a declared pregnant worker is determined that takes the dose equivalent to the embryo/fetus into account.

EPM Response: NRC regulations specify requirements for monitoring the declared pregnant woman. Therefore, any related workplace monitoring is based on potential exposures to the worker and not the embryo/fetus. If a worker declares her pregnancy and monitoring is required, then dose assessment to the embryo/fetus would be performed based on the mother's intake. As indicated previously, evaluations concluded that internal monitoring for declared pregnant workers is not required. Accordingly, dose to the embryo/fetus from intakes will not be performed. However, if the need to perform internal monitoring for declared pregnant worker is indicated, dose to the embryo/fetus would be determined by following guidance provided in Regulatory Guide 8.36 and ICRP Publication 88 as discussed in the May 2017 NRC Staff Periodic Review of Regulatory Guide 8.36, which states the following:

> *Calculating the radiation dose to the embryo/fetus from internally deposited radionuclides requires quantitative information about maternal radionuclide intake, placental transfer and kinetics, and embryo/fetus radionuclide concentrations. The methodology used in RG 8.36 relies mainly on the guidance provided by the International Commission on Radiological Protection (ICRP 30), "Limits for the Intake of Radionuclides by Workers," and NUREG/CR-5631, Revision 1, "Contributions of Maternal Radionuclide Burdens to Prenatal Radiation Dose-Interim Recommendations (1992).*

This guidance is outdated. There are more up-to-date models for estimating dose to the embryo/fetus than the ones listed in this RG, such as ICRP Publication 88, "Doses to the Embryo and Fetus from Intakes of Radionuclides by the Mother," corrected version May

2002. In addition, guidance is provided in ICRP Publication 73, "Radiological Protection and Safety in Medicine" (paragraphs 76 and 77), and in Publication 75, "General Principles for the Radiation Protection of Workers" (paragraph 124).

Changes to the RPP regarding dose to the embryo/fetus were previously discussed.

SECTION 11.6: CONTAMINATION AND CONTROL PROGRAM
The licensee provided a description of its contamination control program (DP 11.6, RPP 10.2-10.4, 12, 13).

Please provide the following information: 1) A description (e.g., maps) of restricted areas established at site, and 2) the types and frequencies of contamination surveys for restricted and contaminated areas (NUREG-1757, Section 17.3.1.6 suggests a matrix or tabular form).

EPM Response: Three figures from Appendix K-2 of the DP have been annotated to show where restricted areas would be located based on the 60% design of the groundwater treatment facilities. These figures will be added to Section 8 of the RPP. Restricted area designations may be modified based on operational experience.

Proposed revisions to the RPP are included in Attachment 4.

SECTION 11.7: INSTRUMENT PROGRAM
The licensee provided a description of its instrument program (DP 11. 7; RPP 7).

Please provide the following information:
1) A description of the method used to estimate Minimal detectable concentration (MDC) or Minimal detectable activity (MDA) (at 95% confidence level) for each type of radiation to be detected and expected radionuclide mixtures (i.e., provide a specific calculation methodology taking into account surface efficiency, etc.). A reference to a desk instruction is not sufficient. See, for example, NRC discussion of minimum detectable concentrations in ADAMS Accession Nos. ML 18072A029 and ML 15295A045,

EPM Response: The MDC is calculated by the following equation:

$$MDC = 3 + 3.29\frac{\sqrt{R_b T_s (1 + {}^{T_s}/_{T_b})}}{E \times T_s}$$

Where,

- R_b is background count rate (counts/minute)
- T_s is sample count time (minutes)
- T_b is background count time (minutes)
- E is instrument efficiency (counts/disintegration)

This equation is equivalent to Eq 3-12 of NUREG-1507. The surface efficiency is taken into account in the determination of the instrument efficiency. The following surface efficiencies factors are used in the development of the instrument efficiency:

- Alpha emitters – 0.25
- Beta emitters – 0.5
- Gamma emitters – 1.0

A surface efficiency factor is not applied to measurements of wipe sample or air samples.

Section 7.5 of the RPP will be clarified to discuss information related to calculating the MDC as discussed above. Proposed revisions to the RPP are included in Attachment 4.

Please note that the two referenced documents refer to in-situ uranium mining operations. There are significant differences between the Cimarron project and such in-situ uranium recovery operations. For an in-situ mining operation:

1. The natural uranium ore body is in secular equilibrium with the entire decay chain progeny.
2. A chemical is injected into the ore body to solubilize the uranium which is then extracted by pumping out the solution to recover the uranium. Other progeny are also extracted in this process.

In the Cimarron project, contaminated groundwater will be extracted, and the low enriched uranium will be removed by an ion exchange process. A portion of the clean groundwater will then be reinjected into the well field. The licensed material (low enriched uranium) was processed prior to receipt by the facility; it does not contain the entire decay chain of progeny. The response to the NRC's request for information related to Section 12.2 (below) identifies the few daughters of uranium that will be present in the licensed material at the site.

2) A description of instrument storage, calibration, and maintenance facilities for instruments used in field surveys, including onsite facilities used for laboratory analyses of samples collected during surveys. A reference to a desk instruction is not sufficient,

EPM Response: Portable survey instruments will be stored in the Instrument Room located in the Western Area Treatment Facility. This includes portable instruments for direct measurements as well as the smear counter(s). No analytical laboratory instrumentation is used at the Site. Laboratory analyses are performed by vendors at off-site laboratories. The floor plan for the WATF building is provided in Appendix K-2 of the DP on drawing K-EPM-DWG-A-100.

CERT currently only replaces batteries, mylar windows, and probe cables. Instrument calibration and maintenance is performed by a qualified vendor. Instrument operation and use are addressed only in procedures; desk instructions are no longer used for this purpose. Proposed revisions to Section 7.5 of the RPP are included in Attachment 4.

3) Section 7.4 of the RPP provides quality assurance (QA) procedures for laboratory instrumentation. Please provide QA procedures for other instruments used in the radiation protection program. A reference to a desk instruction is not sufficient.

> **EPM Response:** As stated previously, the Site does not use laboratory instrumentation and relies on vendors to provide these analytical services. Section 7.4 of the RPP will be revised to remove reference to QA for laboratory instrumentation.
>
> Table 7.1 of the RPP provides a list of the radiological instruments used for the project. Calibration and maintenance of these instruments will be done by an off-site qualified vendor. Procedures are provided for operation and use of instruments at the site. Smears and air samples will be counted on the Ludlum Model 3030E.
>
> Proposed revisions to Section 7.4 of the RPP are included in Attachment 4.
>
> The following procedures are provided in Enclosure C as examples related to the information requested:
> RP-101, "Operation and Use of the Ludlum Model 12 with 44-9 Detector"
> RP-102, "Operation and Use of the Ludlum Model 19"
> RP-103, "Operation and Use of the Ludlum Model 2221 with 44-10 Detector"
> RP-104, "Operation and Use of the Ludlum Model 2360 with 43-93 Detector"
> RP-105, "Operation and Use of the Ludlum Model 3030E"
> RP-106, "Air Sample Collection Using the RADECO AVS-281 Air Sampler"

SECTION 11.9: HEALTH PHYSICS AUDITS, INSPECTIONS, AND RECORDKEEPING

The licensee provided a description of its Health Physics Audits and Recordkeeping program (DP 11.9; RPP 5).

However, there are no details of the Health Physics Audits and Recordkeeping Program in Section 11.9 of the DP or Section 5 of the RPP. The licensee should provide information consistent with the guidance in Section 17.3.3 of NUREG-1757, Vol. 1, Rev. 2. In addition, the licensee should submit its Quality Assurance Program Plan (QAPP) as part of its application.

> **EPM Response:** A general description of the annual radiation protection program review required by 10 CFR 20.1101(c) is discussed in Section 5.2 of the RPP. The RPP commits to relying on NRC guidance for conducting these reviews, including a specific reference to Appendix L of NUREG-1556, Volume 7. NRC Regional inspections of the Cimarron radiation protection program have confirmed compliance with this requirement.
>
> Section 5.3 of the RPP discusses surveillances, which are observations of activities being performed. Surveillances are performed by or under the direction of the Quality Assurance Coordinator and/or the RSO.

Section 5.4 provides a discussion on records of audits and surveillances. This section will be revised to identify the minimum information that must be included in audit and surveillance records. Proposed revisions to Sections 5.3 and 5.4 of the RPP are included in Attachment 4.

SECTION 12: ENVIRONMENTAL MONITORING AND CONTROL
General Comment:
If information relied on for compliance with NRC regulations is contained in procedures or other references (as opposed to the DP itself), it is important to include specific revision numbers for the documents. If a key document has multiple revisions, it can cause confusion in the implementation of the DP, particularly if there is turn over in site staff. It can also lead to the site and the NRC inspectors not having a common understanding of what the licensee is committed to doing. One example of such a document is the RPP. The DP contains a draft version of a revision to this document, but the final version is not provided, and a specific revision number does not appear to be cited in the DP. Please provide the final RPP

EPM Response: CERT performs groundwater sampling and other activities at the Site. The RPP is an operational document that is periodically revised to ensure it is current. NRC Region IV routinely inspects the radiation protection program, including the RPP and procedures.

To respond to previous NRC requests for information and to ensure DP commitments are fully addressed, Revision 4 of the RPP was developed and submitted for NRC review and approval as an appendix to the DP. Currently shown in draft form, Revision 4 will be finalized following acceptance of these responses to the supplemental information requests and any additional requests for additional information from the NRC. CERT has provided markups of affected pages to the RPP resulting from the responses provided in Attachment 4. The final version of RPP, Rev. 4, will be resubmitted to the NRC following completion of NRC review and acceptance of CERT responses.

After Revision 4 to the RPP is approved by the NRC, license condition 27(e) will govern future changes to the RPP. License condition 27(e) reads:

e. The licensee is authorized to make certain changes to the NRC-approved Decommissioning Plan (DP) and Radiation Protection Plan (RPP) without NRC's approval, if these changes are consistent with the ALARA principle and the decommissioning process. All changes shall be approved by the Cimarron ALARA Committee, subject to the following:

1. The licensee may, without prior NRC approval, and subject to the specified in Parts 2 and 3 of this condition:

a. Make changes in the facility or process, as presented in the NRC-approved DP and RPP;

b. Conduct tests or experiments not present in the NRC-approved DP applicable license conditions.

2. The licensee shall not be required to file an application for an amendment to the license when the following conditions are satisfied:

a. The change, test, or experiment does not conflict with requirements specifically stated in the license (excluding those aspects addressed in Part 1 of this condition), or impair the licensee's ability to meet all applicable NRC regulations;

b. There is no degradation in safety or environmental commitments addressed in the NRC-approved DP or RPP, or have a significant adverse effect on the quality of the work, the remediation objectives, or health and safety; and

c. The change, test, or experiment is consistent with the conclusions of actions analyzed in the Environmental Assessment (dated July 29,1999) and Safety Evaluation Report (dated August 20, 1999).

3. If any of these conditions are not met for the change, test, or experiment under consideration, the licensee is required to submit a license amendment application for NRC review and approval. The licensee's determinations as to whether the above conditions are met will be made by the facility's ALARA committee. All such determinations shall be documented. The licensee shall provide in an annual report to NRC, a description of all changes, tests, and experiments made or conducted pursuant to this condition, including a summary of the safety and environmental evaluation of each such action. As part of this annual report, the licensee shall include any DP or RPP pages revised pursuant to this condition. The records shall be retained until license termination. The retained records shall include written safety and environmental evaluations, made by the ALARA committee, that provide the basis for determining whether or not the conditions are met.

4. Radiation protection program procedures or revisions to these procedures do not require review and approval by the ALARA Committee, but do require review and approval by the Radiation Safety Officer.

CERT has implemented a process to ensure that this license condition is satisfied for any changes to the NRC-approved RPP or NRC-approved DP. All changes are submitted to the NRC annually as required by this license condition, so including a specific revision number in the DP would create a conflict with future revisions made in accordance with license condition 27(e).

Radiation protection procedures must comply with the RPP and DP. Both documents set licensee policy for compliance with applicable regulations. Instructions for implementing these commitments is provide in procedures. The NRC has inspected the RPP and implementing procedures throughout CERT's tenure as licensee and consistently found the radiation protection program to comply with applicable regulations and license requirements. CERT plans to continue to operate its program in this manner.

SECTION 12.1: ENVIRONMENTAL ALARA EVALUATION

Section 17.4.1 of NUREG-1757 Vol. 1 Rev. 2 described the information to be provided in a decommissioning plan for the environmental ALARA evaluation program, but the DP does not appear to contain all the requested information.

• The DP does not appear to include a description of the ALARA goals for effluent control.

EPM Response: In addition to specifying compliance with USNRC effluent release limits, CERT will comply with the requirements of the OPDES permit (Appendix H) for uranium discharges to the liquid effluent streams. This limits the uranium discharges to 30 micrograms/L which is equivalent to approximately 10% to 16% of the limits specified in Appendix B of 10 CFR 20 for uranium (depending on the enrichment level). This is effectively serves as an ALARA goal for liquid releases. Section 12.1 of the DP will be revised to clarify that the OPDES limit, referred to as “the MCL,” serves as the ALARA goal for the site. Proposed revisions to the DP are included in Attachment 3.

Section 5.7.1 of the DP provides a discussion of radiological impacts of the facility and notes that the potential for airborne contamination is unlikely because radioactively contaminated materials are either water, moist resin, or wet biomass. However, airborne radioactive contamination may be encountered in the form of a solid, liquid or particulates suspended in air. In accordance with the RPP, proper personnel practices and engineering controls will mitigate onsite and offsite impacts due to airborne radioactive contamination. Air sampling will be performed as indicated previously

• The DP also does not appear to include a detailed description of the procedures, engineering controls, and process controls to maintain doses ALARA.

EPM Response: Sections 8.3.2 and 8.3.3 of the DP provide a description of the ion exchange and biodenitrification systems, respectively. Section 12.3 discusses effluent controls. These discussions provide information on the engineering and process controls being designed into the systems. Only water at concentrations less than the OPDES permit limits for uranium may be released. Processed water above this level will be reprocessed as discussed in the DP. Drawing P-115 Sheet 1 in Appendix K-3 of the DP shows that effluent from the polishing vessel can be routed back to the lead vessel for further processing, if needed. Proposed revisions to 12.2 of the DP, clarifying that processed water exceeding the MCL will be reprocessed, are included in Attachment 3.

• Section 12.1 of the DP refers to the RPP, but the RPP does not appear to have the above noted information either. Additionally, the RPP refers to RP-10 "ALARA Program" as describing how the ALARA program will be implemented. If the information in RP-10 is intended to be used to demonstrate compliance with NRC regulations, it should be provided to the NRC staff and the RPP should cite a particular revision number.

EPM Response: Procedure RP-10 describes in detail the responsibilities and functions of the ALARA Committee. This information is provided in summary form in Section 10.2 of the DP. The ALARA Committee has responsibility to review and approve all aspects of facility operations to assure that ALARA is an essential part of the operations.

RP-10 has been provided in Enclosure D as information to supplement this response. The reference to RP-10 in the RPP does not specifically include a revision number as that information is irrelevant. All revisions of RP-10 must comply with the RPP and the RPP can only be changed in accordance with condition 27(e) of the license.

SECTION 12.2: EFFLUENT MONITORING

Section 17.4.2 of NUREG-1757 Vol. 1 Rev. 2 described the information to be provided in a decommissioning plan on the effluent monitoring program for the NRC staff to be able to fully understand how the effluent monitoring program will be implemented and conducted. The DP does not appear to contain all of the requested information.

• Section 12.2 includes information on the expected maximum concentration of uranium (i.e., the MCL), but it does not include information on the concentration of other radionuclides and whether any other radionuclides are present at levels *above* background.

EPM Response: The only licensed material that will be handled in the groundwater treatment systems is enriched uranium which was processed through the enrichment facilities. The uranium isotopes present are:

- U-234 (no progeny)
- U-235 + Th-231 in secular equilibrium
- U-238 + Th-234 and Pa-234m in secular equilibrium

The activity distribution of the uranium isotopes for 2% enriched uranium is:

- U-234 – 68.3%
- U-235 – 3.7%
- U-238 – 28.1%

Other than the progeny noted above, any other progeny from the U-235 and U-238 decay chains will be from natural radionuclides in the background.

Tc-99 is also be present at the Site at concentrations below the NRC license criterion. Processing of Tc-99 and expected effluent concentrations is addressed in the response to information requested on Section 8.6 of the DP.

• A justification that the sample ports provide representative samples was not provided.

EPM Response: Sampling ports are located on the piping between each influent tank and the ion exchange treatment system between the, between the lead, lag, and polishing resin vessels, and after the polishing resin vessel. Sampling ports are also installed on the piping between the effluent tanks and their outfall structures. Sampling procedures will be developed during 90% design. Those procedures will specify that a minimum volume of water be purged from the sample port prior to collection of the sample. This will ensure that the sample is representative of the water flowing through the piping at the time the sample was collected. This is a component of the "SNM Inventory Control Plan" discussed in Section 11.10 of the DP.

• Section 12.2 does not include a description of the environmental monitoring recording and reporting procedures.

EPM Response: Section 15.3 of the RPP addresses "Quality Control in Sampling." The RPP states, "Sample collection, preservation, shipping, and analysis shall be conducted in accordance with the site-specific Sampling and Analysis Plan and associated procedures. Data review, reporting, and management will be conducted in accordance with Quality Assurance Implementing Procedure, QAIP-17.1, "Data Management Procedure." Section 15.4 of the RPP requires that environmental monitoring results be reported to the NRC within 30 days of the completion of data review.

The Quality Assurance Program Plan (Rev. 4) is included as Enclosure E.

• The description of the quality assurance program for effluent monitoring in Section 12.2 is minimal.

EPM Response: Section 12.2 of the DP commits to following the guidance of Regulatory Guide 4.15 for the Quality Assurance Program Plan. Section 14.5 addresses "Quality Control for Environmental Sampling and Analysis." Environmental monitoring is also covered in Section 15 of the RPP.

Quality Assurance requirements are generally addressed in Section 14 of the DP. The Quality Assurance Program Plan (Rev. 4) is included as Enclosure E.

SECTION 12.3: EFFLUENT CONTROL

Section 17.4.3 of NUREG-1757 Vol. 1 Rev. 2 described the information to be provided in a decommissioning plan for the effluent control. The DP does not appear to contain all of the requested information.

• Section 12.2 does not appear to contain a summary of the action levels and a description of the actions to be taken if a limit is exceeded.

EPM Response: As noted previously, only water at concentrations less than the OPDES permit limits for uranium may be released. Processed water above this level will be reprocessed as discussed in the DP. Drawing P-115 Sheet 1 in Appendix K-3 of the DP shows that effluent from the polishing vessel can be routed back to the lead vessel for further processing, if needed. The MCL serves as the action level and is referenced in Section 12.2 of the DP. Proposed revisions to Section 12.2 of the DP, clarifying that processed water exceeding the MCL will be reprocessed, are included in Attachment 3.

• Section 12.2 does not appear to include a summary of the estimates of doses to the public from effluents and a description of the method used to estimate these public doses per 10 CFR

20.1302. This estimation of doses should account for all radionuclides that are present above background. Note this is related to a previous comment on Section 8.6.

> **EPM Response:** In compliance with 10 CFR 20.1302(b)(2) "Compliance with dose limits for individual members of the public," demonstration that the dose limits to the public are met will be based on measurements of effluent streams at the point of release. This will demonstrate that the dose estimate will be a fraction of the dose limit. Proposed revisions to Section 12.3 of the DP, reflecting the need to demonstrate compliance with this requirement, are included in Attachment 3.

SECTION 13.1: SOLID RADIOACTIVE WASTE

Section 17.5.1 of NUREG-1757 Vol. 1 Rev. 2 described the information to be provided in a decommissioning plan related to solid radioactive waste. However, the following information does not appear to be in the DP:

• Section 13.1 of the DP provided some information on limits for the concentration of U-235 in the resin. However, the expected concentrations and other radionuclides in the resin is not provided. Additionally, Section 13.1 and Section 5 note that the resin will be blended to meet the disposal facility waste acceptance criteria (WAC). It is not clear what criteria in the WAC this blending is being performed to meet (e.g., homogenization or to meet concentration limits).

> **EPM Response:** The limiting conditions for the disposal of the spent resin (LLRW) are both the concentration and homogeneity of the fissile isotope (U-235) established by the transportation regulations for shipment as fissile exempt material. The WAC requirements for the potential disposal sites are higher than the transportation limits. The concentration of the uranium isotopes present in the LLRW waste streams (i.e. spent resin waste) will be established in order to meet all of the transportation regulations for the transport of the waste. The response above for Section 12.2 provides information on the distribution of the uranium isotopes present which depends on the uranium enrichment level.
>
> Groundwater processed from UP1 and UP2 may contain Tc-99. Ion exchange resin used to remove uranium from groundwater recovered from these areas, and biomass generated in the treatment process used to remove nitrate from groundwater recovered from these areas, will be analyzed to determine the concentration of Tc-99 that may accumulate in these materials. This information will be included in the shipping manifest and used for demonstrating compliance with the WAC for the radioactive waste disposal site. In addition, the analysis of both spent resin and biomass for Tc-99 will be added to Table 8-3b, In-Process Monitoring, of the DP. Proposed revisions to the DP are included in Attachment 3.

• The DP does not appear to include the expected volumes of contaminated materials such as gloves, disposable sampling devices or contaminated piping or equipment that will be generated. The DP also does not appear to have information on the approximate expected concentrations of radionuclides on this waste.

> **EPM Response:** The nature of the planned operation will not generate large volumes of this form of waste. Normal conditions of the facilities will maintain operational areas in a clean area status. The primary volume of LLRW that will be generated is that associated with the spent resin. Other volumes of LLRW anticipated are expected to be a fraction of the spent resin volume, on the order of 10% to 15% of the resin volume. The concentration of uranium associated with such potentially contaminated materials is expected to be non-detectable. This waste will be handled and disposed of in the same manner as the spent resin.

• The DP does not address if any volumetrically contaminated waste is expected.

> **EPM Response:** The solid waste expected to be encountered during decommissioning are discussed in Section 13.1 of the DP: Section 13.1.1 discusses spent anion resin and Section 13.1.2 discusses potentially contaminated material. If other volumetrically contaminated material (e.g., sediment or biomass containing detectable concentrations of uranium or Tc-99) is identified, it will be disposed of at a facility which is licensed or permitted to accept that waste. The estimated volume of biomass was included in the cost estimate provided in Section 16 of the DP; it is not known how much, if any, other volumetrically contaminated material will be generated by the water remediation processes.

• The DP does not seem to contain the name and location of the disposal facility that the licensee intends to use for each solid radioactive waste type. CERTs prior response to RAI EA-10 indicates that this information will be contained in Sections 8.3.1, 8.3.2, 8.4.3, 8.6.3, and 8.6.5, but this information does not appear to be in these sections either. The NRC staff wants to know whether there is a contractual obligation for a recipient of the solid radioactive waste.

> **EPM Response:** Two disposal sites are currently available for Cimarron Site radioactive waste; Energy*Solutions* in Utah and Waste Control Specialists in Texas. Contractual arrangements have not yet been made with either of these disposal sites. Proposed revisions to Section 13 of the DP, identifying these sites as potential disposal facilities, are included in Attachment 3. Disposal site information is not relevant to Sections 8.3, "Groundwater Treatment," 8.4, "Treated Water Injection," or 8.6, "In-Process Monitoring."

RADIATION PROTECTION PLAN

General Comment

Throughout the RPP, the licensee used the term "RSO or designee". Please provide the qualifications of the designee (e.g., a qualified health physicist).

> **EPM Response:** The description of the RSO in section 3.2 of the RPP includes the following statement, "The RSO is given specific authority to implement and manage the licensee's radiation protection program, either directly or through qualified individuals who are designated in writing as having authority to exercise specific functions.

When designating an individual to perform certain responsibilities, the RSO considers the designated individual's education, health physics training, and specialized knowledge:

- Education: A Bachelors' degree in the physical sciences, industrial hygiene or engineering from an accredited college or university or an equivalent combination of training and relevant experience in radiological protection. Two years of relevant experience are considered equivalent to 1 year of academic study;

- Health physics experience: At least 1 year of work experience in applied health physics, industrial hygiene or similar work relevant to radiological hazards associated with site remediation. This experience should involve actually working with radiation detection and measurement equipment, not simply administrative or "desk" work; and

- Specialized knowledge: A thorough knowledge of the proper application and use of all health physics equipment used for the radionuclides present at the site, the chemical and analytical procedures used for radiological sampling and monitoring, and methodologies used to calculate personnel exposure to the radionuclides present at the site. The individual must have the appropriate specialized knowledge to perform the designated responsibility.

The RPP requires individuals to be designated in writing. The RSO designates specific individuals through desk instructions, which are updated at least annually or through email for short duration designations.

Proposed revisions to Section 3.0 of the RPP, including a discussion of the education, health physics training, and specialized knowledge required for RSO designees, are included in Attachment 4.

1) The licensee stated (RPP 10.1): "If air sample data indicates a measured level greater than 40 DAC-hours in any shift or operation, whichever is shorter in time duration, the RSO or designee shall conduct an investigation and take corrective actions to reduce airborne contamination levels."

The values for Derived Air Concentration (DAC) in 10 CFR Part 20, Appendix B, Table 1, are derived for various radionuclides and their translocation classification (D, W, or Y).

Please provide the DAC values, and their technical bases, for determining compliance with 10 CFR Part 20.

Please discuss how compliance is assessed against 10 CFR 20.1201(e) for soluble uranium.

EPM Response: The licensed activity only involves withdrawing groundwater to remove the uranium from it and then return a portion of the water to the aquifer. The radionuclides of concern are those associated with low-enriched purified uranium. The DAC for class Y of 2E-11 μCi/mL is used at the site.

Low concentrations of Tc-99 from processing recycled uranium have been detected at the Site. Considering the license termination limit of 3,790 pCi/L, any potential intake would be small fractions of the ALI and total quantities of groundwater processed at this concentration in a year would not require air sampling or internal monitoring. As noted above, the average concentration of Tc-99 that is expected to be encountered is significantly less than the license termination limit.

Assuming an individual consumed the water from the well containing the highest concentration of uranium (Burial Area 1, 3,000 μg/L), an individual would need to drink more than 3 liters of groundwater or inhale 250 times the ALI in one week to have an intake of 10 mg. Drinking of groundwater is never permitted at the site. Potential intake from inhalation is at levels that are inconceivable at the site. Proposed revisions to the RPP to include potential intake calculations were described above.

2) The licensee stated (RPP 10.1): "Air sample collection media shall be appropriate to address the radionuclide mixture(s) present." Please provide the expected radionuclide mixtures and the air sampling media to be used. Please discuss how internal exposure is determined for mixtures of radionuclides (10 CFR 20.1204(g)).

EPM Response: CERT plans to use laminated glass type filters for air sampling. Mixtures of radionuclides have been previously discussed. Internal monitoring and dose assessment was discussed previously. The radionuclides source term of concern is low-enriched uranium.

3) The licensee stated (RPP 10.6.5): "Action levels will be developed that will include specific action levels (i.e., specific projected or actual airborne radioactive material concentration levels) for assigning respiratory protection, collecting bioassay samples, and stopping work." Please provide a list of all airborne action levels developed, actions taken when they are exceeded, and their technical bases.

EPM Response: Please see responses to supplemental information requested on Section 11.1, Air Sampling Program.

4) The licensee stated (RPP 10.6.6): "Minimum detectable activities (MDAs) based on various sample count times will be calculated and used to determine the sample volume needed to detect less than 10% DAC for 4% enriched uranium." Please provide the methodology for calculating the MDA for airborne samples as well as the MDA for each specific radionuclide that may be collected in air samples. Please also address how the potential for the burial of radionuclides within the filter media is assessed when determining filter efficiency (refer to NUREG-1400, "Air Sampling in the Workplace", Section 6.2).

EPM Response: Please see the response to Section 11.7. The same methodology will be applied for MDA calculations. The MDA cannot be provided in this response as it is calculated based upon the background detected by the instrument but must be below the trigger value of 1% DAC. The enrichment referenced will be corrected to 3% for internal consistency in the Decommissioning Plan. Proposed revisions to Section 11 of the DP are included in Attachment 3.

External Exposure Determination
The licensee stated (RPP 6.1): "Area radiation monitoring was established (see Section 10.5 of the RPP) to confirm the results of this evaluation." Please provide the NRC staff a summary of area radiation monitoring results since 2006.

EPM Response: The facility described in the DP has not yet been built and area radiation dosimeters have not been deployed. Area dosimeters locations will be identified when the groundwater treatment system design is 90% complete and will be deployed after the groundwater treatment system is constructed. The current area radiation dosimeters are placed in the site office building which will no longer be used when groundwater treatment facilities are constructed.

The site office includes two Radioactive Materials Areas. Dosimeter results indicate background levels of radiation; between 20 and 30 mrem deep dose equivalent per calendar quarter. Dosimeters began to be placed in areas of potential exposure to licensed material in anticipation of groundwater treatability testing in 2013. Enclosure F is a table presenting summary dosimeter results since License SNM-928 was transferred to the CERT.

Summation of Internal and External Exposures
The licensee stated (RPP 6.1): "Personnel monitoring has not been performed since 2006 because there was no potential to receive a dose that would require monitoring under 10 CFR 20.1502. During the design of groundwater extraction and treatment systems, new work activities, such as groundwater processing, were evaluated to determine if they may result in exposure requiring personnel monitoring." Please provide the analysis that resulted in this conclusion and whether this analysis is current.

EPM Response: The CERT has been the licensee for the Cimarron site since February 2011. Radiological hazards encountered at the site are limited to low concentrations of low-enrichment uranium in groundwater. Activities have typically involved groundwater sampling, limited treatability testing, and a trenching pilot test. None of these activities resulted in the potential for dose to an individual that would require personnel monitoring. Field dose rate measurements consistently measure background exposure rates less than 15 μR/hr.

During 60% design of the groundwater treatment system, a calculation was performed to determine the radiation levels surrounding a resin vessel array used to extract uranium contamination from groundwater at the Cimarron Site. The results of this calculation represent the highest anticipated dose rate an individual would be subject to while in the close vicinity of a uranium treatment train. The highest anticipated dose rate in the vicinity of a uranium treatment train is calculated to be 0.024 mrem/hr. CERT plans to re-evaluate this calculation during final design reviews. As previously discussed, CERT concluded that personnel dosimetry would not be required.

Proposed revisions to the RPP to include the external dose calculations were described above.

Facility Radiation Surveys
Release Criteria

The licensee stated (RPP 13.3) that " ... surveys will be performed and documented by qualified individuals." Please provide the qualifications of a "qualified individual".

EPM Response: CERT has implemented a task qualification process for individuals authorized to perform and document radiological surveys. Documented in a desk instruction approved by the RSO, the task qualification documents on-the-job training provided by a qualified instructor (designated by the RSO) to an individual regarding proper operation and source checking of radiological survey instrumentation identified in a task qualification table/checklist. Successful completion of the task qualification process demonstrates an adequate working knowledge of these instruments that allows the task qualified Individual to perform job coverage surveys for radiation levels (dose rate) and contamination levels within specified limits.

Section 3.0 of the RPP will be revised to discuss the task qualification process. Proposed revisions to Section 3.0 of the RPP, discussing the task qualification process, are included in Attachment 4.

2.7.6 Groundwater Models

Groundwater flow models for the Western Alluvial Area and BA1 were initially developed by ENSR Corporation, and submitted to NRC in *Groundwater Flow Modeling Report*, (ENSR 2006). Those flow models were revised in 2013 and again in 2016, based on information obtained from additional COC delineation and aquifer testing performed in 2013 and additional groundwater assessment performed in 2014. The groundwater flow models incorporate area-specific lithologic and hydraulic detail to describe groundwater gradients and flows and assist in determining the locations and probable production of groundwater from groundwater extraction technologies such as groundwater recovery wells and groundwater extraction trenches. The BA1 and Western Alluvial models were updated with new geologic and hydrogeologic data in 2016. In addition, the Western Alluvial model was expanded to include a larger area in the Western Uplands. Details regarding updates to model input data, calibration, and evaluation were presented in the *2016 Groundwater Flow Model Update* (Burns & McDonnell, 2017) included as Appendix M. As discussed in Section 8.2.4, ~~T~~the groundwater flow models were revised again in 2018 to incorporate revisions to remediation component locations, quantities, and flow rates.

Burial Area #1

The model domain for BA1 is shown on Figure 2-12. There are twelve layers in the model. This complex model layering system setup was initially described in the *2006 Groundwater Flow Modeling Report* (ENSR, 2006b). Flow into the model domain is from recharge both from upgradient and from precipitation, and general head boundaries and flow out of the model is to the Cimarron River. Figure 2-12 also shows the simulated potentiometric surface based on static groundwater elevations (i.e., not influenced by extraction or injection).

Western Alluvial Area

The model domain for the Western Alluvial Area (WAA) is shown on Figure 2-13. The original model domain was expanded eastward to address remedial alternatives in the entire area of the nitrate plume as defined by the 10-mg/L isoconcentration contour; it therefore covers a larger area than the 2006 groundwater model. The WAA model domain includes two layers: Layer 1 represents the alluvium and Layer 2 represents the underlying bedrock. Flow into the model domain is from recharge and general head boundaries and groundwater flow out of the model is to the river. Figure 2-13 also shows the simulated potentiometric surface based on static groundwater elevations (i.e., not influenced by extraction or injection).

The two freshwater ponds (reservoirs) on the Site are located in Subarea B. Subarea B was released for unrestricted use in License Amendment 13, issued April 13, 1996.

The Cimarron River is located along the northern boundary of the Site. Annual environmental monitoring continues to demonstrate that the Cimarron River is not impacted by any of the COCs associated with the Site.

3.5 GROUNDWATER

Groundwater is the only environmental medium for which decommissioning is required to obtain unrestricted release of the Site. This section lists the groundwater assessments that have been performed for the Site and presents the current extent of impact for all COCs in groundwater at the Site.

The NRC Criterion for the Site is 180 picoCuries per liter (pCi/L) total uranium, derived from a risk-based concentration, and stipulated in License Condition 27(c).

Groundwater in several areas of the Site contains two non-radiological COCs: nitrate and fluoride. For uranium and fluoride, the criteria to achieve an unrestricted release from the DEQ are the EPA MCLs for drinking water. The MCLs are 30 μg/L for uranium and 4 mg/L for fluoride. Because nitrate is present at concentrations above the MCL due at least in part to the use of fertilizer, DEQ has designated a value of 22.9 mg/L as the State Criterion, based on analysis of samples from monitor wells located upgradient of processing or disposal activities. The State Criterion for nitrate in the process building area is 52 mg/L.

As detailed in Section 2.7.4, groundwater in the vicinity of BA1 originates from infiltration in the area of the former disposal trenches and Sandstone B from upgradient. Groundwater in Sandstone B enters the Transition Zone of the floodplain alluvium and subsequently enters the sandy alluvial material. In general, groundwater uranium impacts at concentrations greater than the MCL are observed in BA1 in Sandstone B, the Transition Zone, and the floodplain alluvium (see Section 3.5.3 below).

Groundwater in the Western Upland and the Western Alluvium also originates as precipitation that infiltrates into the shallow groundwater unit recharge zones and flows into Sandstone A (see Section 2.7.4). In the Western Upland, the 1206 Drainage (west of Monitor wells 1400, 1354, 1352, etc.) and a smaller drainage to the northeast (east of Monitor Wells 1397, 1340, and 1396) act as local drains for groundwater in Sandstone A, resulting in groundwater base flow discharging from Sandstone A into Transition Zone sediments deposited within these drainages. Groundwater discharged into these drainages also becomes surface water.

In general, groundwater nitrate impacts at concentrations greater than the State Criterion are observed in the Western Area (Western Alluvium and Western Uplands) in Sandstone A, Sandstone B, the Transition Zone, and the floodplain alluvium (see Section 3.5.3 below). Likewise, groundwater fluoride and uranium impacts at concentrations greater than the MCL are observed in Sandstone A, Sandstone B, the Transition Zone, and the floodplain alluvium (see Section 3.5.3 below).

3.5.1 Submittals Addressing Groundwater Assessment

Numerous groundwater assessment efforts have been performed at the Site. The following is a list of reports on groundwater assessment activities.

- April 17, 2002, *Former Burial Area #1 Groundwater Assessment Work Plan*, Cimarron Corporation
- September 24, 2002, *Tc-99 Site Impact Evaluation and Proposed Groundwater Assessment Work Plan*, Chase Environmental Group
- December 12, 2002, *Well 1319 Area Groundwater Assessment Work Plan*, Cimarron Corporation
- January 29, 2003, *Burial Area #1 Ground Assessment Report*, Cimarron Corporation
- December 30, 2003, *Draft Tc-99 Groundwater Assessment Report*, Chase Environmental Group
- December 30, 2003, *Assessment Report for Well 1319 Area,* Cimarron Corporation
- August 10, 2005, *Site-Wide Groundwater Assessment Review,* Cimarron Corporation
- November 5, 2005, *Refined Conceptual Site Model,* ENSR International
- October 19, 2006, *Conceptual Site Model (Revision- 01),* ENSR International
- October 23, 2006, *Groundwater Flow Modeling Report,* ENSR International
- March 3, 2013, *Pneumatic Slug Testing Memorandum,* Burns & McDonnell
- March 15, 2013, *Hydrogeological Pilot Test Report,* Burns & McDonnell
- January 6, 2014, *Groundwater Flow Modeling Report,* Burns & McDonnell
- July 22, 2014, *Hydrogeological Testing Memorandum,* Burns & McDonnell
- May 8, 2015, *Report on 2014 Design Investigation,* Burns & McDonnell
- July 5, 2016, *Distribution Coefficient Determination for the Cimarron Site,* EPM
- January 25, 2017, *Groundwater Flow Model Update,* Burns & McDonnell

3.5.3 Current Extent of COCs in Groundwater

The 2015 *Cimarron Facility Decommissioning Plan* presented data from the 2015 groundwater assessment sampling event. In some areas, COC concentrations appeared to be anomalously low in 2015, whereas in other areas, COC concentrations appeared to be consistent with or slightly higher than previous data. NRC requested that groundwater data be evaluated for evidence of seasonal variability, as well as to determine if changes in COC concentrations were related to changes in groundwater elevation.

Quarterly collection of groundwater samples from 44 monitor wells was begun in the first quarter of 2016. Samples were collected from wells screened in all three sandstone units, in transition zone material in the WAA and BA1, and in alluvial material in the WAA and BA1. Data from 2011 through the Fourth Quarter of 2016 were evaluated, and the evaluation results were presented in *2016 Groundwater Evaluation* (Burns & McDonnell, 2017). The evaluation concluded that there is no relationship between either season or groundwater elevation and COC concentrations. This evaluation was updated in *2017 Groundwater Evaluation* (Burns & McDonnell, 2018), yielding the same conclusion.

It is necessary to minimize the potential for individual data points to exercise undue influence on the estimated concentrations of COCs to treatment trains. Consequently, the decision was made to determine the concentration of each COC at each location at the 95% upper confidence level, based on data obtained from 2011 through the second quarter of 2017. For locations for which the 95% upper confidence level was greater than the maximum concentration, the maximum concentration was used. For locations for which less than 4 data points were available, the average concentration was used.

Figures 3-1 through 3-4 present isoconcentration contours (isopleths) for each COC, based on the results of these calculated concentrations. Figure 3-1 presents an isopleth map for nitrate in the Western portion of the Site~~A~~. Figure 3-2 presents an isopleth map for fluoride in the Western portion of the Site. Figure 3-3 presents an isopleth map for uranium in the Western portion of the Site. ~~Figure 3-4 presents an isopleth map for uranium in BA1.~~ As shown on Figure 3-3, representative uranium concentrations in the Western Area range from 0.63 to 875 µg/L. The average representative uranium concentration in the Western Area is 47.2 µg/L and the maximum and average representative uranium concentrations within each aquifer in the Western Area are as follows:

Aquifer	Maximum Representative Uranium Concentration (µg/L)	Average Representative Uranium Concentration (µg/L)
Alluvium	178	53.8
Sandstone A	875	43.4
Sandstone B	38.0	5.43
Transition Zone	527	333

The maximum and average representative uranium concentrations within each Western remediation area exceeding the NRC Criterion of 180 pCi/L are as follows:

Remediation Area	Maximum Representative Uranium Concentration (µg/L)	Average Representative Uranium Concentration (µg/L)
WAA U>DCGL	178	85.0
1206-NORTH	527	333
WU-BA3	875	203

The remediation areas with uranium concentrations exceeding the NRC Criterion are shown on Figure 3-3, as are iso-concentration contours depicting the magnitude and extent of uranium contamination. Representative uranium concentrations for each Western Area monitor well are also presented in table form on Figure 3-3, and wells with concentrations exceeding NRC or DEQ criteria are indicated on the table via color coding.

Figure 3-4 presents an isopleth map for uranium in BA1. As shown on the figure, representative uranium concentrations in BA1 range from 1.24 to 3516 µg/L. The average representative uranium concentration in BA1 is 412 µg/L and the maximum and average representative uranium concentrations within each aquifer in BA1 is as follows:

Aquifer	Maximum Representative Uranium Concentration (µg/L)	Average Representative Uranium Concentration (µg/L)
Alluvium	3516	277
Sandstone B	2589	307
Transition Zone	2975	857

The maximum and average representative uranium concentrations within each BA1 remediation area exceeding the NRC Criterion of 180 pCi/L are as follows:

Remediation Area	Maximum Representative Uranium Concentration (µg/L)	Average Representative Uranium Concentration (µg/L)

BA1-A	2975	599
BA1-B	3516	388

The remediation areas with uranium concentrations exceeding the NRC Criterion are shown on Figure 3-4, as are iso-concentration contours depicting the magnitude and extent of uranium contamination. Representative uranium concentrations for each BA1 monitor well are also presented in table form on Figure 3-4, and wells with concentrations exceeding NRC or DEQ criteria are indicated on the table via color coding.

~~Additionally,~~ Attachment 2.1 of the Basis of Design, included in Appendix L, presents the maximum, average, and 95% UCL (if available) nitrate, fluoride, and uranium groundwater concentrations for site monitor wells, based on results generated by groundwater monitoring events conducted from 2011 through the Second Quarter 2017. This attachment also presents the representative nitrate, fluoride, and uranium groundwater concentrations used as the basis for remediation design. The protocols and methods used to determine representative COC groundwater concentrations are described in Attachment L.

The maximum and average representative nitrate concentrations observed in the Western Area are 1,006 mg/L (Monitor Well 1385) and 71.5 mg/L, respectively (see Figure 3-1). The maximum and average fluoride concentrations observed in Western Area are 48.9 mg/L (Monitor Well 1313) and 3.1 mg/L, respectively (see Figure 3-2). The maximum and average uranium concentrations observed in Western Area are 875 µg/L (Monitor Well 1351) and 47.2 µg/L, respectively. The maximum and average uranium concentrations observed in BA1 are 3,516 µg/L (Monitor Well TMW-13) and 412 µg/L, respectively.

Estimated average influent uranium concentrations and remediation system design flow rates (refer to Section 8.2.4 for details) were used to estimate the mass of uranium that will be recovered from each remediation area exceeding the NRC remediation criterion, from the time remediation begins until the NRC remediation criterion is achieved (refer to Section 9.3 for details regarding remediation durations and schedule). The estimated mass that will be recovered from each remediation area exceeding the NRC remediation criterion is as follows:

- BA1-A: 393 kg (achieved in 150 months)
- BA1-B: 137 kg (achieved in 45 months)
- WAA U>DCGL: 16.2 kg (achieved in 38 months)

- 1206-NORTH: 0.99 kg (achieved in 5 months)
- WU-BA3: 11.5 -kg (achieved in 49 months)

  Note: the uranium mass removed from WU-BA3 will be flushed from WU-BA3 via treated water injection and subsequently recovered by extraction trench GETR-WU-02 located in the 1206-NORTH remediation area.

The values used to calculate uranium enrichment must be as accurate as reasonably achievable to estimate the mass of U-235 that may accumulate in ion exchange resin vessels during groundwater treatment. Isotopic analysis performed prior to 2016 consisted of alpha spectroscopic analysis of isotopic activity. At the relatively low uranium concentrations that exist throughout much of the area requiring remediation, the uncertainty associated with the calculated enrichment is high. In estimating enrichment values for uranium, the "mean plus 2-sigma" enrichment value for all data obtained at each location was calculated. Due to the high uncertainty associated with isotopic ***activity*** analysis, this calculation method resulted in an over-estimation of enrichment values for the groundwater treatment system influent streams.

In December 2016, groundwater samples were collected from multiple locations to obtain a data set spanning the variability of uranium enrichment and concentration that occurs across the Site. Samples were analyzed for isotopic activity by alpha spectroscopy and for isotopic mass concentration by inductively coupled plasma – mass spectroscopy (ICP-MS). The data was evaluated to determine which method would provide the most accurate isotopic results at low uranium concentrations. The result of this evaluation was reported in a technical memorandum entitled, "*Analysis of Analytical Method for Uranium Enrichment Determination*" (Enercon Services, 2017). The evaluation conclusively demonstrated that ICP-MS analysis produces isotopic results with far less uncertainty at low concentrations.

Groundwater samples were then collected from 197 monitor wells for isotopic analysis by ICP-MS during the Second Quarter of 2017. Groundwater samples were collected from all monitor wells located in areas where groundwater will be extracted for treatment, as well as areas from which groundwater will be driven to extraction components by the injection of treated water. Samples were analyzed for mass concentration of the U-235 and U-238 isotopes only, because the mass of U-234 at the low enrichment levels encountered at the Site is negligible (less than 0.05% of the total uranium mass), In-process groundwater elevation data will be used to maximize the driving head from areas of upland COC impact toward groundwater extraction features, while minimizing the potential for contaminant displacement to areas outside the boundaries of capture zones.

Discharge Monitoring

Groundwater treatment discharge monitoring will be conducted to demonstrate compliance with requirements set forth in the OPDES Discharge Permit and 10 CFR 20.2001.

The EPA has determined that 900 pCi/L represents the beta emitter activity concentration that equates to the maximum contaminant level (MCL) of 4 millirem per year for beta emitters, while NRC has determined that a beta activity concentration of 3,790 pCi/L equates to the 4 millirem per year MCL. The primary contributor to the beta activity detected in groundwater at the Site is believed to be Tc-99 and Tc-99 has only been reported above laboratory detection limits in groundwater samples collected in the Western Area. Because the expected WATF influent Tc-99 concentration (466 pCi/L) is expected to comply with the lower of the two drinking water standards described above, Tc-99 was not addressed in the OPDES permit application; consequently, OPDES Discharge Permit OK0100510 (included as Appendix H) does not stipulate a limit for Tc-99, nor does it require monitoring for Tc-99.

To evaluate compliance with 10 CFR 20.2001, it was conservatively assumed that no Tc-99 will be removed during treatment of the WATF influent groundwater prior to discharge. The effluent limits for uranium and Tc-99 listed in 10 CFR Part 20 Appendix B, are 3×10^{-7} and 6×10^{-5} µCi/mL, respectively. These concentration limits are equivalent to 300 and 60,000 pCi/L, respectively. The concentration of uranium in WATF effluent is expected to be below laboratory detection limits; however, for the purposes of evaluating compliance with 10 CFR 20.2001, a WATF effluent uranium concentration of 5 pCi/L was conservatively assumed. Using the average Tc-99 concentration of 466 pCi/L presented above, a unity rule analysis for estimated activity concentrations of uranium and Tc-99 in effluent discharges is as follows:

$$\sum = \left(5\frac{pCi}{L} \Big/ 300\frac{pCi}{L} \right) + \left(466\frac{pCi}{L} \Big/ 60{,}000\frac{pCi}{L} \right) = \mathbf{0.024}$$

Uranium and Tc-99 concentrations in effluent will be a small fraction of NRC-stipulated effluent limits.

The flow rate to each outfall will be recorded, and samples of treated water being discharged via each outfall will be collected for laboratory analysis, on a bi-weekly basis. Discharge monitoring reports will report this data to DEQ on a monthly basis in accordance with the OPDES discharge permit. Parameters and locations for in-process discharge monitoring are presented in Table 8-3c.

8.6.4 Groundwater Remediation Monitoring

Concentrations of groundwater COCs requiring remediation will be monitored to evaluate progress toward remediation goals and to determine when remediation within a given area or area should be discontinued and post-remediation groundwater monitoring should begin. In-process monitor wells used to evaluate remediation progress are the same as those previously specified for groundwater extraction and injection performance monitoring. Locations of the in-process monitor wells are depicted on Figures 8-8 and 8-9. Table 8-2 lists the wells by remediation area and identifies the COCs to be analyzed for groundwater samples collected from each well.

In-process monitoring of COC concentrations in groundwater will consist of the sampling and analysis of select monitor wells in each subarea. Monitoring COC concentrations within each remediation area will provide the information needed to adjust remediation process parameters, primarily extraction and injection flow rates, assess progress toward remediation goals, evaluate when operation of specific wells or trenches can be discontinued, and determine when remediation in a specific area can cease and post-remediation monitoring can begin. Post-remediation groundwater monitoring is addressed in more detail in Section 8.8, Post-Remediation Groundwater Monitoring.

In-process groundwater monitoring will provide several years of data which can be used to evaluate the rate of decline of COC concentrations in groundwater. Section 8.1.7 states that post-remediation monitoring will begin when at least three consecutive months of in-process monitoring data shows that all wells yield uranium concentrations below 180 pCi/L. However, evaluation of in-process monitoring data may indicate that treatment should continue to reduce the risk of exceeding those criteria during post-remediation monitoring.

In addition to evaluating remedial progress, in-process groundwater monitoring results will be used to assess the effectiveness of specific remediation components in each area. Based on the results, groundwater extraction and injection system operations may be adjusted to focus efforts on areas with higher levels of impact, maximizing COC mass recovery and concentration

License Condition 27(e) also specifies the evaluation the ALARA Committee must perform to determine if a change to tests, the Decommissioning Plan, or the Radiation Protection Plan require NRC approval. If not, the ALARA Committee can approve the change without NRC approval. The ALARA Committee sets ALARA dose goals for the Cimarron site.

This Plan restricts the concentration of licensed material in effluents generated during decommissioning to less than the MCL. A proposed change to the decommissioning process that could impact effluent concentrations would require the ALARA Committee to review the proposed change in accordance with License Condition 27(e). The change evaluation will be documented and maintained on site for review during regulatory inspections.

ALARA Committee meeting agenda and minutes, change evaluations and approvals of changes, and proposed and/or approved modifications of ALARA goals and processes, are distributed to all members of the ALARA Committee. Consequently, management remains fully informed of all ALARA issues associated with the decommissioning and release of the Site.

12.2 EFFLUENT MONITORING

The extent and concentration of both licensed material (i.e., uranium) and non-radiological contaminants of concern (i.e., nitrate and fluoride) have been established as described in Section 3.5.3 of this decommissioning plan.

Once groundwater remediation has begun, effluents will consist of extracted groundwater containing less than the MCL for each COC and Tc-99, a radionuclide associated with contamination resulting from the enrichment process that may be present in groundwater. Effluents will be discharged to the Cimarron River via DEQ-permitted Outfalls 001 and 002. The locations of the two outfalls are shown on Drawing C002 in Appendix J-2. Samples of the discharge will be collected from sampling ports installed on the pipeline discharging from Effluent Tanks TK-102 (discharging to Outfall 001) and TK-202 (discharging to Outfall 002). Discharge sample ports are collected near the effluent tanks because they are located outside of the 100-year floodplain and are not subject to flooding. Samples will be analyzed in accordance with OPDES Permit No. OK0100510.

Sample collection frequency, compositing, and analytical methods are stipulated in the OPDES permit. A procedure for discharge sampling will be prepared in accordance with the Site quality assurance program and added to the DEQ-approved Sampling and Analysis Plan.

Samples will be collected twice monthly and analyzed for uranium, nitrate, and fluoride. Samples will be analyzed for pH, uranium, nitrate, ~~and~~ fluoride, and Tc-99. The minimum quantification limit for nitrate is 50 µg/L; samples will be analyzed for nitrate by method EPA 353.2, which has a detection limit of 17 µg/L. The minimum quantification limit for fluoride is 1,000 µg/L; samples will be analyzed for fluoride by method EPA 300.0, which has a detection limit of 66 µg/L. There is no specified minimum quantification limit for uranium; samples will be analyzed for uranium by method EPA 200.8, which has a detection limit of 0.067 µg/L, which is significantly less than the MCL of 30 µg/L.

The OPDES permit specifies daily maximum concentration limits of 30 µg/L for uranium, 10 µg/L for nitrate, and 10 µg/L for fluoride. The pH of discharged water must be between 6.5 and 9.0 standard units. As stated in Section 8.6.3, because the expected WATF influent Tc-99 concentration is expected to comply with the EPA drinking water standard, which is lower than the NRC standard, Tc-99 was not addressed in the OPDES permit application; consequently, OPDES Discharge Permit OK0100510 (included as Appendix H) does not stipulate a limit for Tc-99, nor does it require monitoring for Tc-99. Section 8.6.3 also includes a demonstration of compliance with NRC effluent criteria stipulated in 10 CFR 20.2001.

The OPDES permit is issued for a five-year period. Should renewal be necessary, during the fifth year, ten samples will be collected each of ten months from each effluent tank's sample port for analysis for manganese, arsenic, chromium, copper, lead mercury, selenium, thallium, zinc, cyanide, and barium. In addition, one surface water sample will be collected the Cimarron River from an upstream location; these samples will be analyzed for mercury and thallium. The OPDES permit requires monthly reporting of flow and analytical results on a monthly discharge monitoring report by the fifteenth day of each month.

The QAPP established for the Site complies with Regulatory Guide 4.15, *Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination) – Effluent Streams and the Environment*. The QAPP has been reviewed during multiple NRC inspections, and will be revised in accordance with OPDES permit requirements, as appropriate.

12.3 EFFLUENT CONTROL

Releases of radioactive material to the environment can occur during groundwater remediation through:

- A leak or leaks in well heads or piping

13.0 RADIOACTIVE WASTE MANAGEMENT

13.1 SOLID RADIOACTIVE WASTE

Solid radioactive waste generated by groundwater remediation activities will fall into one of several categories:

- Spent anion resin
- Potentially contaminated material (e.g., protective clothing, materials, and equipment) used to maintain the systems and process groundwater (i.e., dry active waste, or DAW)
- Contaminated piping and equipment removed from ion exchange treatment systems

Biomass from the biodenitrification system~~s~~ is not anticipated to contain detectable concentrations of uranium; however, it is not known whether the biomass may contain detectable concentrations of Tc-99. Spent anion resin and biomass generated by the biodenitrification system will be analyzed to determine the concentration of Tc-99 that may accumulate in these materials. This information will be included in the shipping manifest and used for demonstrating compliance with the WAC for the radioactive waste disposal site~~ and will not be managed as solid radioactive waste~~. If the biomass contains detectable concentrations of Tc-99, it will be disposed of at a facility which is licensed or permitted to accept that waste.

13.1.1 Spent Anion Resin

Anion resin beds will contain approximately 750 kg resin. Estimates based on concentrations in groundwater indicate that no resin vessel will ever accumulate more than 500 grams of U-235, because as the uranium concentration of influent groundwater declines, the adsorption capacity of the resin declines. Consequently, a single resin vessel will be unable to adsorb sufficient uranium to exceed the U-235 possession limit of 1,200 g. The total mass of U-235 in all treatment trains combined is not expected to exceed 800 grams at any given time. In addition, the processed spent resin will contain less than one-gram U-235 per 2 kg non-fissile material.

The resin processing operation involves blending spent resin with non-fissile material in a ribbon blender. No chemicals will be used, as the non-fissile material will consist of an inorganic absorbent. This will result in uniform distribution of SNM throughout the resin/additive mixture (blended waste) in compliance with transportation requirements. The blended waste will be packaged in 55-gallon drums (or other suitable containers as required).

Table 8-3b
In-Process Treatment System Monitoring
Weekly Sampling for Analysis

Location		Sample ID	pH (on-site)	U-235 & U-238 by EPA 200.8	Nitrate by EPA 353.2	Fluoride by EPA 300.0	Tc-99 by HASL 300
WATF Train 1	Lead Vessel Influent (pre-acid addition)	AE100	X				X
	Lead Vessel Influent (post-acid addition)	AE101	X	X	X	X	
	Lead Vessel Effluent			X			
	Lag Vessel Effluent			X			
	Polish Vessel Effluent		X	X	X		X
	Nitrate System Effluent				X	X	X
WATF Train 2	Lead Vessel Influent (pre-acid addition)	AE150	X				X
	Lead Vessel Influent (post-acid addition)	AE151	X	X	X	X	
	Lead Vessel Effluent			X			
	Lag Vessel Effluent			X			
	Polish Vessel Effluent		X	X	X		X
	Nitrate System Effluent				X	X	X
BA1 Train 3	Lead Vessel Influent (pre-acid addition)	AE 200	X				
	Lead Vessel Influent (post-acid addition)	AE201	X	X			
	Lead Vessel Effluent			X			
	Lag Vessel Effluent			X			
	Polish Vessel Effluent		X	X			

Note: Samples to be collected the first business day of each week.

11.0 RADIATION PROTECTION PROGRAM

Appendix N

The Site Radiation Protection Plan (RPP) establishes radiation protection program requirements that will be implemented during decommissioning (extraction and treatment of uranium-impacted groundwater), specifically related to radiation safety controls and monitoring for workers. The licensee implemented a RPP that was approved by the NRC in Amendment 15 to the Site's license, SNM-928. Since NRC approval, the RPP has been revised in accordance with License Condition 27(e) numerous times, each time reflecting changing conditions at the site. Each year, evaluations performed by the ALARA Committee approving those changes, along with updated versions of the RPP, have been submitted to NRC, and have been reviewed during the numerous NRC inspections conducted since Amendment 15 established the change process. Revision 4 of the RPP is included as ~~Appendix O~~ of this decommissioning plan.

To prepare for the extraction and treatment of uranium-impacted groundwater, several changes have been incorporated into the RPP. The sections "Radioactive Materials Control" and "Contamination Control" have been revised to monitor the extraction and treatment of groundwater containing low levels of licensed radioactive material to prevent the spread of contamination to other areas of the site, which would then require re-characterization and possible additional decommissioning. A section on material control and accountability has been added to the RPP to address radiological hazards associated with this processing. The quantity of special nuclear material being generated will be monitored to maintain compliance with the license possession limit of 1,200 grams of U-235, and to achieve the fissile exemption for all LLRW packaged for transportation and disposal. The section "Environmental Monitoring" addresses the sampling and analysis of environmental media until it is superseded by in-process (and subsequent post-remediation) monitoring.

Radiation protection (RP) procedures have been developed to provide consistent, effective performance of effluent monitoring, contamination control, and dose rate monitoring. RP procedures include the sampling and analysis of influents and effluents to monitor the accumulation of SNM in resins, the sampling of loaded resin and biomass for waste characterization, and the sampling, analysis, handling, storage, manifesting, transportation, and disposal of LLRW.

Operating procedures will address the removal and packaging of spent resin and biomass, replacement of resin, the operation and maintenance of groundwater extraction and injection systems, and the operation and maintenance of the water treatment system(s).

11.1 AIR SAMPLING PROGRAM

The air sampling program for the Cimarron Site is described in the RPP, which requires that air samples be collected whenever airborne radioactivity levels are expected to exceed ~~10%~~ 1% of the Derived Air Concentration (DAC) as listed in Appendix B, Table 1 "Occupational Values" of 10 CFR 20. Considering the types of work activities described in this Decommissioning Plan, airborne suspension of licensed radioactive material is not anticipated to generate airborne radioactivity approaching ~~10%~~ 1% of a DAC. These activities include:

- Construction of water treatment and storage facilities, excavation of trenches for piping and/or utilities, and construction of new roads and improvement of existing roads will all take place in areas where soils have been demonstrated to comply with criteria for unrestricted release. Nearly all of these activities will occur in areas which have been released from License SNM-928. Installation of groundwater extraction and injection wells borings, as well as excavation of groundwater extraction trenches, will occur in two zones:
    - Unsaturated soils which comply with criteria for unrestricted release and will not generate airborne radioactivity during work activities.
    - Saturated soils where groundwater may exceed unrestricted release criteria, but do not have the potential for generating airborne radioactivity during work activities.
- Operation of the groundwater extraction and treatment systems are not expected to generate airborne radioactivity. Operation of treated water discharge and irrigation involves water that complies with drinking water criteria and cannot generate airborne radioactivity.
- Influent and effluent sampling involves the collection of liquid streams, where there is no potential for generation of airborne radioactivity. The same holds true for collection of in-process and post-remediation groundwater sampling.
- Removal and processing of treatment media (resin and biomass) involves the handling and packaging of moist material containing SNM. Due to the moisture of the treatment media, airborne radioactivity is not expected to be generated. Because the treatment media represents the only material for which a potential for exposure exists, the removal and processing of treatment media will be evaluated for potential airborne exposure.

General area air sampling, using low- or high-volume portable air samplers, will be performed throughout the resin unloading and packaging process for at least the first three resin exchanges. Following analysis of the air sample results from each resin exchange, the RSO will determine the need and frequency of additional air sampling. Selection of air samplers is based on the following criteria:

- Sampling heads will be placed in the breathing zone, so the air sample is representative of the air breathed by the individual worker.
- Air samplers typically sample at a rate of approximately 3 - 25 liters per minute for a low-volume sampler to 70 ft^3 per minute for a high-volume sampler. Based on the nature of the low enriched uranium encountered, the detection capability of the air sampling equipment and associated radiological analysis (e.g., sample counting) will be evaluated to determine the total volume of air needed to detect ~~10%~~ 1% of the DAC. The enrichment of the uranium will be based on either the actual enrichment being collected on the resin or a conservative basis (i.e. ~~4%~~ 3%). This calculation will be documented in a site procedure or technical basis document. As the actual enrichment of recovered uranium in each area (i.e., WA or BA1), ~~10%~~ 1% DAC may be recalculated. Minimum collection times will be determined so adequate sensitivities are achieved for a given monitoring period.
- The need for air sampling will be prospectively determined based on the final process system design and potential for generation of airborne radioactivity. Due to the chemical and physical nature of the uranium-bearing media (e.g., water and moist ion exchange resin), minimal, if any, airborne radioactivity is expected to be generated. Engineering and physical controls incorporated into the process equipment design will also be considered in determining the need for air monitoring.
- The frequency of calibration of the flow meters on the air samplers will be based on manufacturers' recommendations (typically annually).
- Action levels will be developed that will include specific action levels (i.e., specific projected or actual airborne radioactive material concentration levels) for assigning respiratory protection, collecting bioassay samples, and stopping work.

Air samples will be counted on-site using existing laboratory bench top scalers (e.g., Ludlum Model 3030 or similar equipment). Minimum detectable activities (MDAs) based on various sample count times will be calculated and used to determine the sample volume needed to detect less than ~~10%~~ 1% DAC for 3% enriched uranium. This information will be documented and used to determine the minimum sampling time required.

11.2 RESPIRATORY PROTECTION

Replace deleted text with Insert 11.2.

~~The RPP states that the need for respiratory protection, for radiological work, is not envisioned at the Cimarron Site. However, if work activities are encountered that could potentially expose workers to 40 or more DAC-hours in a week, respiratory protection will be required. Respiratory protection will also be required for any areas where airborne radioactive material concentrations are expected to exceed 1 DAC. If either of these trigger levels are encountered, the RPP will be revised, and a respiratory protection procedure (or procedures) will be established that includes:~~

- Process controls, engineering controls, or procedures to control concentrations of radioactive materials in air
- Evaluations performed when it is not practical to apply engineering controls or procedures
- Considerations used to demonstrate respiratory protection equipment is appropriate
- Required medical screening and fit testing
- Use, maintenance, and storage of respiratory protection devices
- Respiratory protection training program
- Selection of respiratory protection equipment

11.3 INTERNAL EXPOSURE DETERMINATION

Based on anticipated radiological work involving extraction and treatment of groundwater at the Site, a bioassay program is not warranted. If radiological conditions change or evaluation of the final groundwater processing equipment design indicates that an individual worker could be exposed to 2% of the annual limit on intake in a year, then bioassay will be performed. Bioassay will be performed, whenever a calculated intake of 40 DAC-hours occurred in any one incident based on air sampling data, accident conditions, equipment failure, external contamination, or other conditions. Bioassay sampling will also be performed whenever it is likely that an individual may have received an intake of 10 milligrams of uranium in any one week. ~~In addition to the requirements set forth in the RPP, RP procedures include requirements for how worker intakes are determined:~~

If a bioassay program is necessary, then RP procedures would be implemented to include requirements for determining worker intakes, including the following considerations.

- Using measurements of quantities of radionuclides excreted from or retained in the human body
- By measurements of the concentrations of airborne radioactive materials in the workplace
- For an adult, a minor, and a declared pregnant woman using any combination of the measurements above, as may be necessary

INSERT FOR DP SECTION 11.2:

The need for a respiratory protection for radiological work is not envisioned at the Cimarron Site. Work activities that could potentially expose workers to airborne radioactive material have been evaluated to determine the potential intakes during groundwater treatment and spent resin processing. The evaluation employed the methods discussed in Regulatory Guide 8.25, Rev. 1, "Air Sampling in the Workplace" and NUREG-1400, "Air Sampling in the Workplace." If processes or operations change, then a re-evaluation of potential intakes will be performed to determine the potential intake that could result from these changes. If the potential intake determined from this evaluation is 2% ALI or greater, then the RSO will consider the need to implement a respiratory protection program as discussed in this section.

As discussed in the RPP, respiratory protection measures will be employed when necessary to protect workers from airborne hazards. Groundwater treatment results in the generation of moist treatment media with little potential to generate airborne radioactivity. However, as future conditions change and the RSO or designee determines, through review of field conditions or anticipated work functions, that respiratory protection is required, procedures and controls will be instituted in accordance with the requirements found in 10 CFR 20, Subpart H, “Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas” for radiological hazards and the Code of Federal Regulations Title 29 Part 1910.134 for non-radiological hazards.

If a respiratory protection program is determined to be necessary, the program will be based on guidance provided in Regulatory Guide 8.15, Rev. 1, "Acceptable Programs for Respiratory Protection," and NUREG/CR-0041, Rev. 1, "Manual of Respiratory Protection Against Airborne Radioactive Material."

Respiratory protection will be required in areas where workers could be exposed to 40 or more DAC-hours in a week or where concentrations of airborne radioactive material are expected to exceed 1 DAC. If a respiratory protection program is determined to be needed, then the RPP will be revised, and a respiratory protection procedure (or procedures) will be established that includes:

would

RP procedures also describe how worker intakes are converted into committed effective dose equivalent.

11.4 EXTERNAL EXPOSURE DETERMINATION

Individual monitoring for external exposure is not expected to be required during groundwater extraction and processing and related activities. Passive area radiation monitoring using thermoluminescent dosimeters or optically stimulated luminescent dosimeters will be performed to demonstrate that individuals will not exceed the requirements for individual monitoring provided in the RPP. However, as discussed in the RPP, individual monitoring devices will be assigned if any of the following conditions are encountered or expected to be encountered:

- Any individual is likely to receive, from radiation sources external to the body, a dose in excess of 10% of the 10 CFR Part 20 occupational dose limits in a year.
- Any minor is likely to receive, in 1 year, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem, a lens dose equivalent in excess of 0.15 rem, or a shallow dose equivalent to the skin or the extremities that exceeds 0.5 rem.
- A declared pregnant woman is likely to receive during the entire pregnancy, from radiation sources external to the body, a deep dose equivalent that exceeds 0.1 rem.

would be developed that

In addition to the requirements set forth in the RPP, RP procedures describe the type, range, sensitivity, and accuracy of required of individual-monitoring devices. RP procedures also include a description of the action levels for worker's external exposure, and the technical bases and actions to be taken when they are exceeded.

would

Because non-uniform external radiation fields are not expected to be encountered at the Site, RP procedures do not address the use of extremity and whole-body monitors in these situations. Additionally, procedures do not require audible-alarm dosimeters and pocket dosimeters due to low external dose rates that will be encountered during groundwater processing. Determining external dose from airborne radioactive material is not required as the only radionuclides encountered will be uranium isotopes. Area dosimeters and job-specific surveys will be conducted to determine the need for supplemental personnel monitoring.

11.5 SUMMATION OF INTERNAL AND EXTERNAL EXPOSURE

As provided in the RPP, internal and external doses are summed whenever positive doses are measured. Records of total effective dose equivalent and total organ dose equivalent for monitored workers are maintained by the Trust. RP procedures will be used to document the methodology for

License Condition 27(e) also specifies the evaluation the ALARA Committee must perform to determine if a change to tests, the Decommissioning Plan, or the Radiation Protection Plan require NRC approval. If not, the ALARA Committee can approve the change without NRC approval. The ALARA Committee sets ALARA dose goals for the Cimarron site.

, which serves as the ALARA goal for the site.

This Plan restricts the concentration of licensed material in effluents generated during decommissioning to less than the MCL. A proposed change to the decommissioning process that could impact effluent concentrations would require the ALARA Committee to review the proposed change in accordance with License Condition 27(e). The change evaluation will be documented and maintained on site for review during regulatory inspections.

ALARA Committee meeting agenda and minutes, change evaluations and approvals of changes, and proposed and/or approved modifications of ALARA goals and processes, are distributed to all members of the ALARA Committee.

Only water at concentrations less than the OPDES permit limits for each COC may be released. Processed water above this level will be reprocessed as discussed in the DP. Drawing P-115 Sheet 1 in Appendix K-3 of the DP shows that effluent from the polishing vessel can be routed back to the lead vessel for further processing, if needed.

ALARA issues associated with the d

12.2 EFFLUENT MONITORI

The extent and concentration of both contaminants of concern (i.e., nitrate and fluoride) have been established as described in Section 3.5.3 of this decommissioning plan.

Once groundwater remediation has begun, effluents will consist of extracted groundwater containing less than the MCL for each COC. Effluents will be discharged to the Cimarron River via DEQ-permitted Outfalls 001 and 002. The locations of the two outfalls are shown on Drawing C002 in Appendix J-2. Samples of the discharge will be collected from sampling ports installed on the pipeline discharging from Effluent Tanks TK-102 (discharging to Outfall 001) and TK-202 (discharging to Outfall 002). Discharge sample ports are collected near the effluent tanks because they are located outside of the 100-year floodplain and are not subject to flooding. Samples will be analyzed in accordance with OPDES Permit No. OK0100510.

Sample collection frequency, compositing, and analytical methods are stipulated in the OPDES permit. A procedure for discharge sampling will be prepared in accordance with the Site quality assurance program and added to the DEQ-approved Sampling and Analysis Plan.

Samples will be collected twice monthly and analyzed for uranium, nitrate, and fluoride. Samples will be analyzed for pH, uranium, nitrate, and fluoride. The minimum quantification limit for nitrate off all groundwater extraction pumps. As the treatment system continues to operate, the low-level sensor will then trigger a shut-down of the pumps transferring water to the treatment system. Even if both tanks fail simultaneously (and catastrophically), the maximum volume of impacted water that could be released would be a single volume of the influent tank.

Each uranium treatment train is installed within a shallow containment. If a resin vessel (or a connection to a resin vessel) develops a leak, a conductivity sensor will trigger a shut-down of the pumps transferring groundwater from the influent tank to the treatment train. The maximum volume of the release will therefore be the volume of water in the treatment train. Because most of this water has already been in contact with the resin, the concentration of licensed material in the resin will be significantly less than that of the influent.

There is no release of impacted water to a sewer system, so the requirements of 10 CFR 20.2003 do not apply to this decommissioning operation.

12.4 STORMWATER CONTROL

Stormwater runoff during construction activities has the potential to impact the environment, particularly surface water. As discussed in Section 5.6.13, A Notice of Intent to comply with OPDES General Permit OKR10 was submitted to the DEQ on November 6, 2017. The DEQ authorized the discharge of stormwater in accordance with the general permit in a letter dated June 25, 2018. The SWPPP for the full-scale construction project will be prepared after the 90% design is complete and RAIs have been received and reviewed.

BMPs will be installed, and corrective measures will be conducted and documented in accordance with SWPPP requirements. A Notice of Termination for the OPDES General Permit will be submitted following establishment of a minimum 70% coverage with perennial vegetation.

Any detectable concentrations of licensed material in water released as an effluent will be evaluated to demonstrate compliance with the dose limits set forth in 10 CFR 20,1302.

Water processed from UP1 and UP2 may contain Tc-99. Resin used to process this water will be analyzed to determine the concentration of Tc-99 collected on that resin. This information will be included in the shipping manifest and used for demonstrating compliance with the WAC for the radioactive waste disposal site.

Initially, a sample collected from each drum will be analyzed for isotopic concentration. The collection of multiple samples from a single batch provides the data needed to assess the homogeneity of the mixture. Analytical data will be the basis for shipping papers and will provide the data needed to document that transportation and disposal criteria have been met. Table 8-3d presents the sample identification and analytical method for samples of processed resin.

Four 55-gallon drums will be loaded onto a pallet and the drums will be strapped together. Pallets of filled drums will be labeled and placed in a designated area within the Secured Storage Facility located east of the WATF Building (see Drawing C-110, Appendix K-1) pending receipt of analytical results. The Secured Storage Facility is a Metal Building with a single roll-up door that will have removable bollards to additionally restrict access to the interior of the facility (see Drawings A-170 [Appendix K-6] and C-110 [Appendix K-1], respectively).

Palleted drums will be stored in the secured storage facility until enough drums have been stored to constitute a full consignment. The spent resin mixture will then be shipped by common carrier to a licensed disposal facility for disposal as Class A, fissile excepted, low level radioactive waste.

Discussions have been held with two potential waste disposal sites; EnergySolutions in Utah and Waste Control Specialists in Texas to confirm that the packaged waste will conform with their respective WAC.

The ble[…] site. The blended waste will comply with the following requirements:

- The SNM will be uniformly distributed throughout the matrix of the resin, a hydrocarbon material. This material is considered soil-like but is not a SiO_2 matrix.
- The waste form will be in containers which will be disposed at the licensed disposal site in accordance with license requirements for containerized waste for the disposal site.

~~Discussions have been held with the proposed waste disposal site to confirm that the packaged waste will conform with the WAC.~~ The analysis demonstrating that a potential criticality condition related to the transportation or disposal of the spent resin mixture is not credible has been incorporated into Appendix O.

13.1.2 Potentially Contaminated Material

Gloves, small diameter tubing, and other materials which may become contaminated during groundwater processing are not expected to absorb sufficient uranium to exceed surface contamination limits. However, since these cannot be surveyed practically to demonstrate this,

TABLE OF CONTENTS

All radiation protection procedures shall be reviewed and approved by the RSO.

All radiation protection procedures shall by reviewed by the QAC or designee for conformance with quality assurance program requirements.

All radiation protection procedures shall be controlled in accordance with regulatory requirements and the Quality Assurance Program Plan.

3.7 Desk Instructions

Desk instructions may be developed and implemented to provide a reference guide on specific topics that help the user implement various aspects of the RPP. Desk instructions may be written to address use of specific radiation survey instrumentation or details associated with electronic survey form completion. Desk Instructions are issued by the RSO or designee and expire 12 months after approval. Desk Instructions may be renewed at additional 12 month increments.

3.8 Notifications and Reports

Notifications and reports shall be made in accordance with the requirements of 10 CFR 19, 10 CFR 20 and 10 CFR 70.

See insert for section 3.9

INSERT FOR RPP SECTION 3.9

3.9 RSO Designees and Task Qualification

3.9.1 Prior to designating an individual, the RSO should consider the following:

3.9.1.1 Education: The designated individual should have a Bachelors' degree in the physical sciences, industrial hygiene or engineering from an accredited college or university or an equivalent combination of training and relevant experience in radiological protection. Two years of relevant experience are considered equivalent to 1 year of academic study.

3.9.1.2 Health physics experience: The designated individual should have at least 1 year of work experience in applied health physics, industrial hygiene or similar work relevant to radiological hazards associated with site remediation. This experience should involve actually working with radiation detection and measurement equipment, not simply administrative or "desk" work.

3.9.1.3 Specialized knowledge: The designated individual should have a thorough knowledge of the proper application and use of all health physics equipment used for the radionuclides present at the site, the chemical and analytical procedures used for radiological sampling and monitoring, and methodologies used to calculate personnel exposure to the radionuclides present at the site. The individual must have the appropriate specialized knowledge to perform the designated responsibility.

3.9.2 Designated individuals may be qualified to perform specific tasks approved by the RSO. A modified "systematic approach to training" is employed to qualify individuals on specific tasks. Task qualifications must be documented and include the following:

3.9.2.1 Verify that the selected individual has sufficient experience (e.g., related technical experience, such as environmental remediation, industrial hygiene, use of scientific instruments, etc.), education (including physical science and math), and prior training (related to the specific task, which may include electronic equipment use and handling, computer applications, etc.).

3.9.2.2 Learning objectives based on the procedural requirements to perform the task.

3.9.2.3 On-the-job training including performance terminal objectives that the individual must satisfy through performance, simulation, or discussion, Each performance terminal objective should include the behavior being evaluated (e.g., task being performed), conditions associated with the task, standards that must be met (e.g., applicable procedures), and the steps necessary to perform the specific task.

3.9.2.4 Task qualifications are typically valid for 12 months and may be extended with refresher training and RSO or designee approval.

5.0 ASSESSMENTS

5.1 Section Overview

Audits and/or surveillances provide a review of decommissioning and radiation protection activities to evaluate compliance with regulatory requirements, license conditions, and the radiation protection plan and procedures. Audits and/or surveillances identify unsatisfactory performance and/or weaknesses in procedures, training, or work practices. The results of all audits and surveillances are reviewed by the ALARA Committee.

5.2 Audits

10 CFR 20.1101(c) requires that a licensee shall, at least annually, review the radiation protection program content and implementation. Various NRC guidance documents (e.g. Appendix L, NUREG-1556, Vol. 7) provide sample forms to assist the licensee in meeting this requirement.

Periodic audits (review of documentation and records), the ALARA Committee review of the RPP and an annual audit modeled on NRC's sample audit form are used to meet this requirement. Periodic audits are conducted, as required, under the Quality Assurance Program Plan. Audits shall be documented, as well as program changes resulting from audit findings or observations.

5.3 Surveillances

Surveillances are observations of activities being performed. Surveillances of Site activities are done by, or under the direction of, the Quality Assurance Coordinator and/or the RSO. The goal of surveillances is to determine whether or not an activity is being performed in accordance with applicable procedures, plans, accepted industry standards, etc. Surveillances shall be documented, as well as program changes resulting from findings or observations made during surveillances. Surveillances are conducted once each calendar quarter at a minimum.

5.4 Records

Records of audits and surveillances are maintained in accordance with the Quality Assurance Program Plan (QAPP).

Audit and surveillance records shall included the following information:

- The date(s) the audit/surveillance was conducted.
- Name of person(s) conducting the audit/surveillance.
- Areas audited/reviewed.
- Audit/surveillance findings, corrective actions, and follow-up.

6.0 PERSONNEL MONITORING

6.1 Individual Monitoring of Occupational Dose

NRC regulation 10 CFR 20.1502 requires the licensee to monitor occupational exposures from both licensed and unlicensed radiation sources. Monitoring is required of any adult likely to receive, in 1 year from sources external to the body, a dose in excess of 10 percent of the Occupational Dose Limits for Adults and/or who are likely to receive, in 1 year, an intake in excess of 10 percent of the applicable annual limit on intake (ALI) in Table 1, Columns 1 and 2, of Appendix B to 10 CFR 20.1001-20.2402. Monitoring for minors is required when they are likely to receive, in 1 year, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem, a lens dose equivalent in excess of 0.15 rem, or a shallow dose equivalent to the skin or the extremities in excess of 0.5 rem and/or likely to receive, in 1 year, a committed effective dose equivalent in excess of 0.1 rem. Monitoring of declared pregnant women is required when they are likely to receive during the entire pregnancy, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem and/or likely to receive during the entire pregnancy, a committed effective dose equivalent in excess of 0.1 rem.

Personnel monitoring has not been performed since 2006 because there was no potential to receive a dose that would require monitoring under 10 CFR 20.1502. During the design of groundwater extraction and treatment systems, new work activities, such as groundwater processing, were evaluated to determine if they may result in exposure requiring personnel monitoring. The threshold dose for personnel monitoring will not be approached; accordingly, neither monitoring workers for external or internal occupational dose is required. Area radiation monitoring was established (see Section 10.5 of the RPP) to confirm the results of this evaluation. Air sampling during spent ion exchange resin handling activities will be performed as discussed in Section 6.5, below, and Section 11.1 of the Decommissioning Plan.

See insert for section 6.1

6.2 Occupational Dose Limits

NRC Regulation 10 CFR 20.1201 establishes a total effective dose equivalent (TEDE) limit and a total organ dose equivalent (TODE) limit for occupationally exposed adults. The TEDE is the sum of the deep dose equivalent (DDE) from external exposures and the committed effective dose equivalent (CEDE) from internal exposures. The TODE is the sum of the DDE and the committed dose equivalent (CDE) to the organ receiving the highest dose. The following annual dose limits apply to all the licensee employees, contractors, and visitors who receive occupational dose at the Cimarron Site.

Occupational dose is defined as the radiation dose an individual receives in a Restricted Area and other work-related radiation dose the person receives. Occupational dose does

INSERT FOR SECTION 6.1:

Two calculations were performed to determine the potential radiological conditions that may be encountered when the groundwater treatment system is operational. One calculation was performed to determine the potential intake from the operations presenting the highest risk of generating airborne radioactive material – resin processing. The other calculation was performed to determine external dose rates from a spent resin vessel. The results of these calculation are provided in Appendix A (EPM028-CALC-001, Potential Intake Calculation) and Appendix B (EPM017-CALC-001, Dose Rate Near Uranium Treatment Train) of the RPP.

These calculations were based on the 60% design of the groundwater treatment system. The potential intake calculation supported the decision that internal monitoring (e.g., bioassay) and respiratory programs were not needed at the Site. This calculation also informed the development of the air sampling program described in Section 10.6 of the RPP. The dose rate calculation supported the decision that personnel dosimetry was not required at the Site.

Both calculations will be reviewed at 90% design, updated, if necessary, and re-evaluated to determine if the RPP should be updated. In addition, periodically through groundwater processing, these supporting calculations will be reviewed to ensure they reflect operational experience and if changes to the RPP are necessary.

pregnant. The Site shall ensure the dose to the embryo/fetus of a declared pregnant woman does not exceed regulatory limits due to occupational dose during the entire pregnancy.

If the internal monitoring for declared pregnant workers is detemined to be necessary, procedures for determining dose to the embryo/fetus will be developed and implemented. Dose to the embryo/fetus will be determined based on guidance provided in Regulatory Guide 8.36 and ICRP Publication 88.

6.8 Summation of Internal and External Dose

Internal and external doses are summed
Procedures will be used to document the
external doses to workers and internal do
embryo/fetus of a declared pregnant worker.

6.9 ALARA Dose Goals

As discussed in Section 4.3, ALARA dose goals will be set if individual monitoring is required. Until such time, the annual Administrative Dose Goals for the Site is effectively 100 mrem TEDE. In cases where Administrative Dose Goals are exceeded without prior authorization, the RSO or designee shall investigate to determine the cause and prepare a written report.

6.10 Personnel Exposure Reports

An annual report of the individual radiation dose received shall be sent to each worker who was issued individual dosimetry and/or was subject to the requirements for monitoring as specified in Section 6.1. When requested by an individual, a written exposure report shall be provided to each such individual within 30 days of the request or within 30 days of exposure determination, whichever is later.

Internal and external doses shall be summed whenever positive doses are measured. The dose to the lens of the eye, skin, and extremities are not included in the summation. Intakes through wounds or skin absorption shall be evaluated and, to the extent practical, accounted for in summation of internal and external doses independent of intakes by ingestion or inhalation.

6.11 Dosimetry Records

Records of individual monitoring shall be kept in accordance with 10 CFR 20.2106 and the Trust QAPP. These records shall be updated at least annually for any radiation monitoring data collected. All radiation exposure records shall use the units curie, rem, rad, or multiples thereof and shall clearly and specifically indicate the quantities (e.g., deep dose equivalent) and units (e.g., rem or mrem) of all recorded values.

Records of total effective dose equivalent and total organ dose equivalent for monitored workers shall be maintained.

Benchtop smear/sample

7.0 RADIATION PROTECTION INSTRUMENTATION

7.1 Calibration

Calibration of radiation monitoring, counting, and air sampling instruments shall be performed in accordance with the manufacturers' recommendation unless otherwise approved by the RSO. These calibrations shall be consistent with regulatory requirements.

The calibration frequency for portable radiation monitoring instruments and portable air sampling equipment shall be at least every 12 months. ~~Semi portable (e.g., continuous air monitors) and fixed (e.g., count room/laboratory instrumentation, portal monitors)~~ instrumentation shall be calibrated at least annually.

7.2 Operation and Response Tests

Operation and response tests of radiation monitoring, counting, and air sampling instruments, shall only be performed by personnel trained in the use of the instrument and following approved procedures. Operation and response tests shall be conducted as required by radiation protection procedures. ~~Desk Instructions may be used to provide guidance on certain aspects of operation and response tests.~~

7.3 Maintenance and Repair

Maintenance and repair of radiation protection instrumentation shall be performed by qualified personnel or an approved vendor. All maintenance and repair shall be documented.

7.4 Quality Control/Quality Assurance

Quality Control (QC) measures for instruments shall be established and maintained to ensure reliability of counting results and sensitivities. ~~Quality Assurance (QA) for laboratory instrumentation shall be proceduralized and consistent, to the extent practicable, with the requirements of USNRC Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment.~~

7.5 Radiation Protection Instrumentation Inventory

Table 7-1 provides a list of equipment available to perform radiological surveys at the Site. Procedures provide implementing requirements for the program and ~~a series of~~

~~desk~~ instructions ~~provide guidance~~ for using specific instruments, including the following information:

- Instrumentation storage, calibration, and maintenance facilities for instruments used in field surveys

See insert for section 7.5

- ~~The method used to estimate the~~ MDC ~~or MDA~~ (at the 95 percent confidence level) for each type of radiation to be detected, as appropriate
- Instrument calibration and quality assurance

Table 7-1

Radiation Protection Instrument List

Make	Model	Probe	Description
Ludlum	Model 12	44-9	Handheld analog ratemeter with a GM pancake-type detector
Ludlum	Model 19	N/A	Gamma micro-R meter (0 to 5000 μR/hr)
Ludlum	2360	43-93	Alpha-Beta Ratemeter, Scaler, and Data Logger with a dual phosphor scintillation probe
Ludlum	3030E	43-10-1	Dual channel, scaler-type sample counter with a dual phosphor detector
Ludlum	2221	44-10	Handheld ratemeter and scaler with an analog display for viewing the count rate with a 2" X 2" NaI(Tl) scintillator
Air Sampling Equipment			
Make	**Model**	**Filter Head**	**Description**
RADEco	AVS-28A	2500-42	Portable, low volume, continuous flow air sampler with a 47 mm diameter open face filter head

INSERT FOR RPP SECTION 7.5

NOTE: The MDC for portable survey instruments is calculated by the following equation:

$$MDC = 3 + 3.29\frac{\sqrt{R_b T_s(1 + {}^{T_s}/_{T_b})}}{E \times T_s}$$

Where,

R_b is background count rate (counts/minute)
T_s is sample count time (minutes)
T_b is background count time (minutes)
E is instrument efficiency (counts/disintegration)

This equation is equivalent to Eq 3-12 of NUREG-1507. The surface efficiency is taken into account in the determination of the instrument efficiency. The following surface efficiencies factors are used in the development of the instrument efficiency:

- Alpha emitters – 0.25
- Beta emitters – 0.5
- Gamma emitters – 1.0

A surface efficiency factor is not applied to measurements of wipe sample or air samples.

See insert for section 8.1

8.0 ACCESS CONTROL

8.1 Section Overview

This section provides the access control requirements for entry into and exit from Restricted Areas (RAs). Access control is designed to ensure that individuals have appropriate qualifications, training, and authorization for entry. Access control requirements are applicable to personnel, contractors and visitors who enter RAs. Restricted Areas are areas within the Site boundary for which access is controlled for the purpose of protecting individuals against undue risk from exposure to radiation and/or radioactive materials.

Restricted Areas will be established based on the potential for accumulating radioactive material greater than ten times the 10 CFR 20 Appendix C quantities or requiring posting as Radiation Areas, High Radiation Area, Contaminated Area, or Airborne Radioactivity Areas.

8.2 Restricted Area Access Controls

Only properly trained or escorted personnel shall be permitted inside any RA. Personnel who enter RAs may be required to wear dosimetry. RAs include Radioactive Materials Areas, Radiation Areas, High Radiation Areas, Contaminated Areas, and Airborne Radioactivity Areas. RAs can be controlled through the use of guards, barriers, fences, signs, gates, or doors.

RA boundaries shall be defined by the use of postings, barriers, walls, tape, ropes, markings, or locked doors. A log of personnel entry and exit to any Restricted Area at the Site will be maintained by the RSO or designee.

8.3 Posting and Labeling Requirements

Posting of areas within each RA shall be performed in accordance with 10 CFR 20 Subpart J. Containers of radioactive materials shall be labeled in accordance with 10 CFR 20.1903. Exceptions to posting requirements found in 10 CFR 20.1903 and exceptions to labeling requirements found in 10 CFR 20.1904 shall be approved by the RSO or designee.

Signs used for posting radiological areas within an RA shall include the wording provided in Table 8-1 when the associated requirements are expected to be encountered or expected to be encountered:

TABLE 8-1

INSERT FOR RPP SECTION 8.1:

The tentative designation of Restricted Areas during initial groundwater treatment are provided in the following figures:
-Figure 8-1: Western Area Secure Storage Facility
-Figure 8-2: Western Area Treatment Facility
-Figure 8-3: Burial Area 1 Treatment Skid

NOTE: These figures are annotated versions of drawings taken from the Decommissioning Plan. Restricted Areas will be periodically reviewed and may be expanded, reduced, or reconfigured based on RSO evaluation of potential exposure to radioactive material. Additional areas may be designated as Restricted Areas if appropriate.

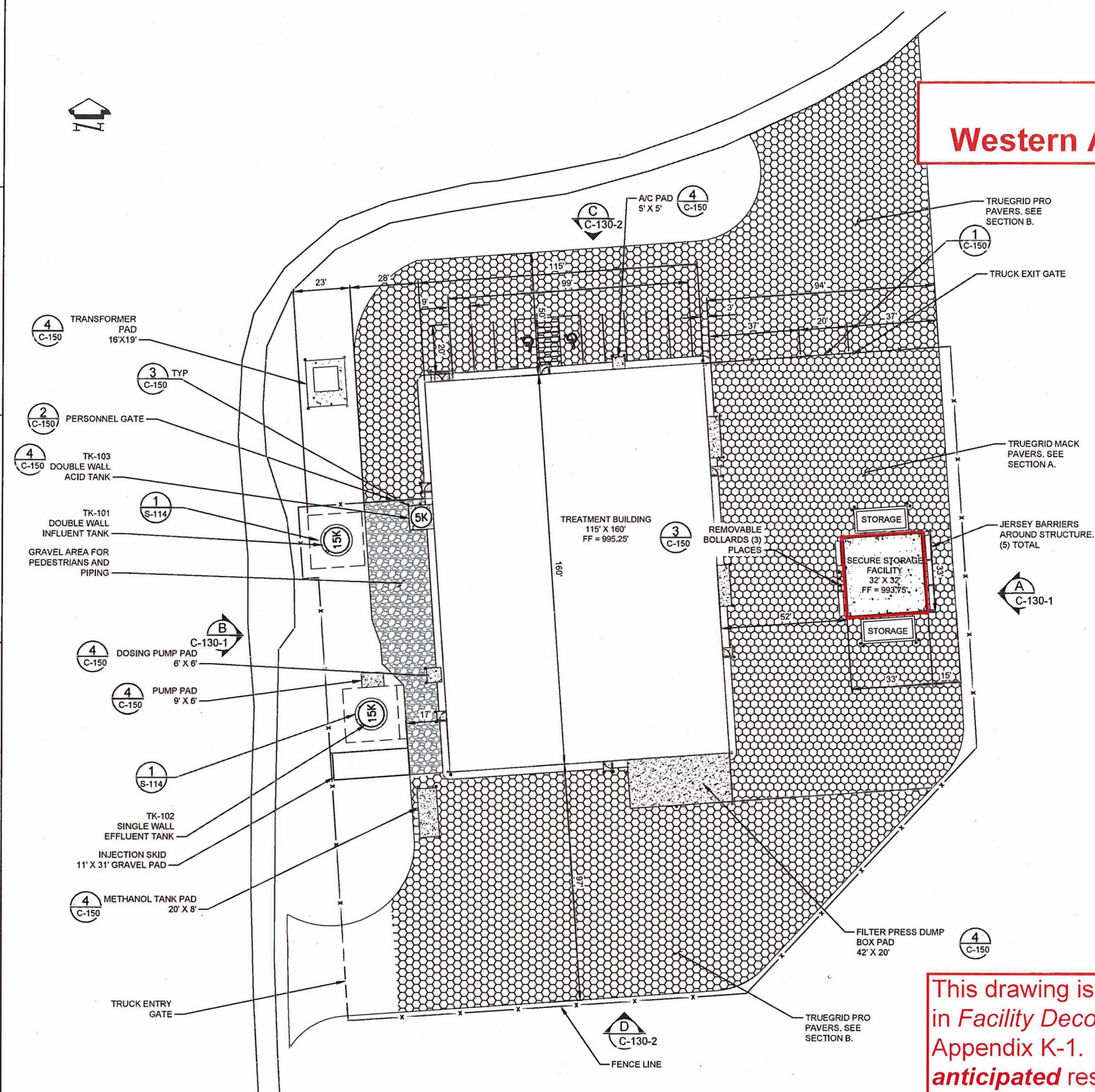
Figure 8-1
Western Area Secure Storage Facility

GENERAL NOTES:

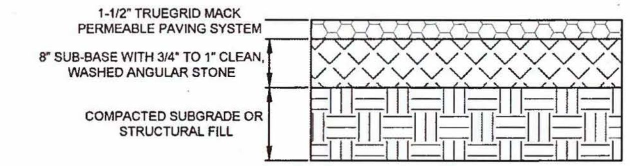
1. SEE K-EPM-DWG-C-112 FOR SITE GRADING PLAN.
2. SEE K-EPM-DWG-C-113 FOR SITE UTILITY PLAN.
3. SEE K-EPM-DWG-S-101 FOR STRUCTURAL NOTES.

TANK INFORMATION NOTES:

1. 15,000 GALLON TANK DIMENSIONS:
   DIAMETER = 10'-6" (126")
   HEIGHT = 24'-0" (288")
2. 5,000 GALLON TANK DIMENSIONS:
   DIAMETER = 8'-6" (102")
   HEIGHT = 18'-0" (216")



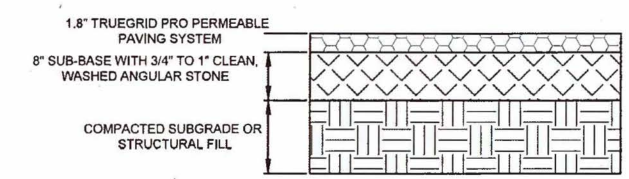
SITE PLAN
SCALE: 1" = 20'-0"



TRUEGRID MACK NOTES:

1. FOLLOW MANUFACTURER'S RECOMMENDATIONS FOR INSTALLATION OF TRUEGRID PRODUCT
2. USE $\frac{3}{8}$" TO $\frac{3}{4}$" AGGREGATE FOR GRAVEL FILL-IN OF GRID.

TRUEGRID MACK PAVING SECTION A
SCALE: NONE



TRUEGRID PRO NOTES:

1. FOLLOW MANUFACTURER'S RECOMMENDATIONS FOR INSTALLATION OF TRUEGRID PRODUCT
2. USE $\frac{3}{8}$" TO $\frac{3}{4}$" AGGREGATE FOR GRAVEL FILL-IN OF GRID.

TRUEGRID PRO PAVING SECTION B
SCALE: NONE

This drawing is an annotated version of a drawing in *Facility Decommissioning Plan - Rev 1 -* Appendix K-1. A red border indicates the ***anticipated*** restricted area. Outside of the Western Area Treatment Facility building, the only location where licensed material will be present is where drums of processed spent resin are stored in the "Secure Storage Facility".

IN-PROCESS
60% DESIGN PACKAGE UPDATE
NOT APPROVED FOR CONSTRUCTION

VEOLIA

VEOLIA
NUCLEAR SOLUTIONS
1355 Columbia Park Trail
Richland, WA 99352
www.nuclearsolutions.veolia.com

NAME	DATE	COMPANY
DRAWN BY D NELSON		VNSFS
ENGINEER D NELSON		VNSFS
CHECKER R RASMUSSEN		VNSFS
PROJECT MANAGER J LUEY		VNS

CIMARRON
PUMP AND TREAT SYSTEM PROJECT

WESTERN AREA
TREATMENT FACILITY
SITE PLAN

SIZE	DWG NO	REV
F	K-EPM-DWG-C-110	B

SCALE AS SHOWN | SHEET 1 OF 1

DWG NO	TITLE
DRAWING TRACEABILITY LIST	

REF NUMBER	TITLE
REFERENCES	
NEXT USED ON	

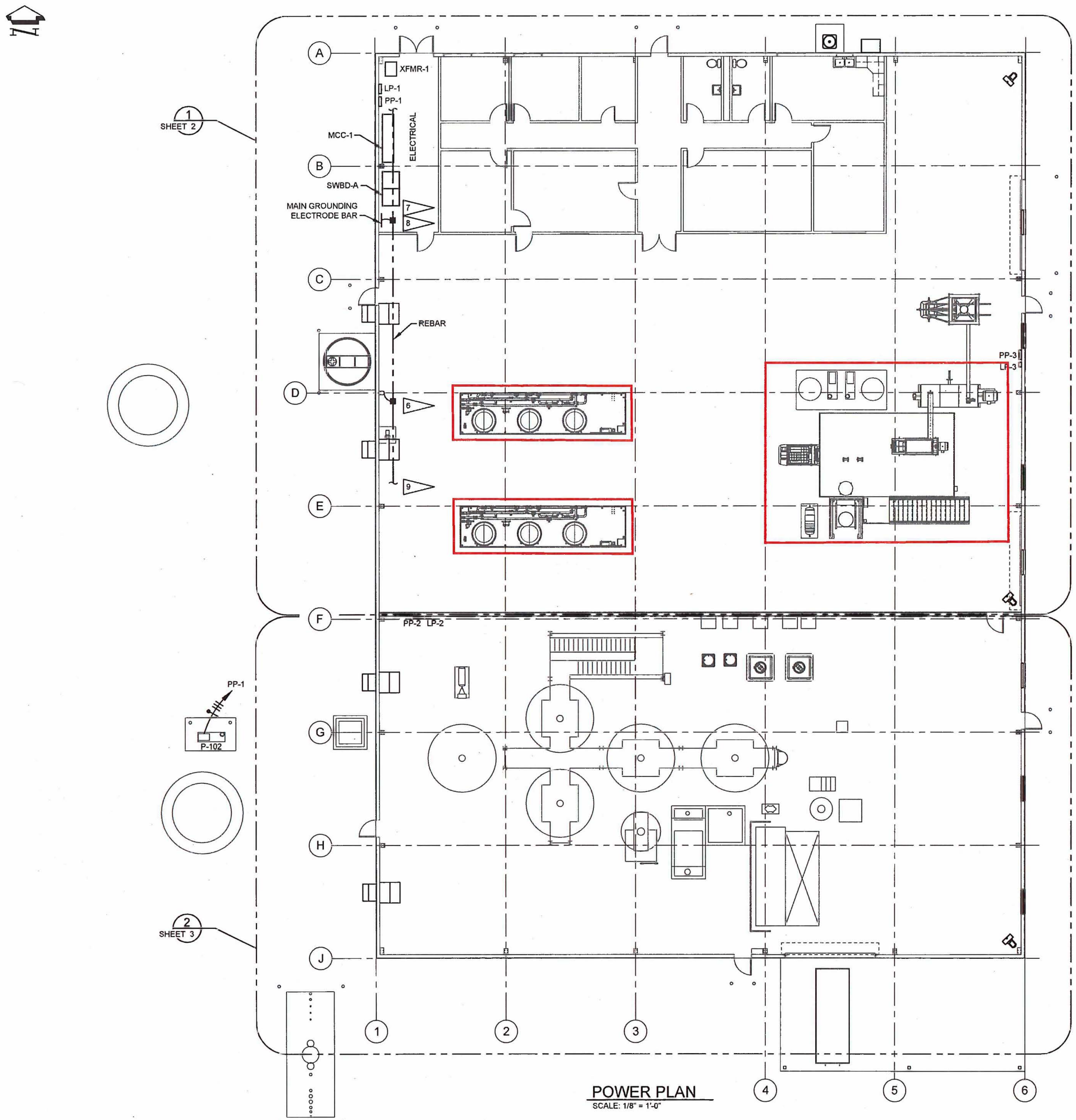
REV	DESCRIPTION	REV BY / DATE	ENGR	COMPANY
REVISIONS				

DWG NO K-EPM-DWG-C-110 | SH 1 OF 1 | REV

GENERAL NOTES:

1. SEE K-EPM-DWG-E-101 FOR SYMBOLS, NOTES, AND ABBREVIATIONS.
2. REFER TO SINGLE LINE DIAGRAM K-EPM-DWG-E-110 FOR CONNECTION DETAILS.
3. ALL GROUNDING SYSTEM CONDUCTORS ARE #4/0 AWG COPPER UNLESS OTHERWISE SPECIFIED.
4. BOND ALL METAL WATER PIPES WITHIN 5 FEET OF SERVICE ENTRANCE TO GROUNDING ELECTRODE SYSTEM.
5. TIE 1/2 INCH MINIMUM DIAMETER EMBEDDED REBAR SECTIONS TOGETHER USING STEEL WIRE TIES OR WELDING TO FORM A CONTINUOUS CONCRETE-ENCASED GROUNDING ELECTRODE.
6. BOND EVERY OTHER STEEL COLUMN TO GROUNDING ELECTRODE SYSTEM.
7. INSTALL MAIN GROUNDING ELECTRODE GROUND BAR IN APPROXIMATE LOCATION SHOWN. CONNECT TO GROUNDING ELECTRODE SYSTEM.
8. PROVIDE 5 FEET OF GROUNDING ELECTRODE CONDUCTOR ABOVE FINISHED FLOOR FOR BONDING TO MAIN GROUNDING ELECTRODE GROUND BAR.
9. REBAR BONDED TOGETHER IN CONCRETE SLAB.
10. POWER CIRCUITS FOR SECURITY LIGHTING AND DETECTION SYSTEMS TO BE DEFINED FOR 90% DESIGN.

Figure 8-2
Western Area Treatment Facility



This drawing is an annotated version of a drawing in *Facility Decommissioning Plan - Rev 1 -* Appendix K-2. A red border indicates the ***anticipated*** restricted areas. In the Western Area Treatment Facility, the only locations where licensed material will be accumulated or can be released during processing is in the fully enclosed Uranium IX Treatment Skids and the spent resin processing facility. The actual restricted area may be expanded, reduced, or reconfigured based on RSO evaluation of potential for exposure to licensed material. Additional areas may be designated as restricted areas if appropriate.

IN-PROCESS
60% DESIGN PACKAGE UPDATE
NOT APPROVED FOR CONSTRUCTION

VEOLIA
VEOLIA NUCLEAR SOLUTIONS
1355 Columbia Park Trail
Richland, WA 99352
www.nuclearsolutions.veolia.com

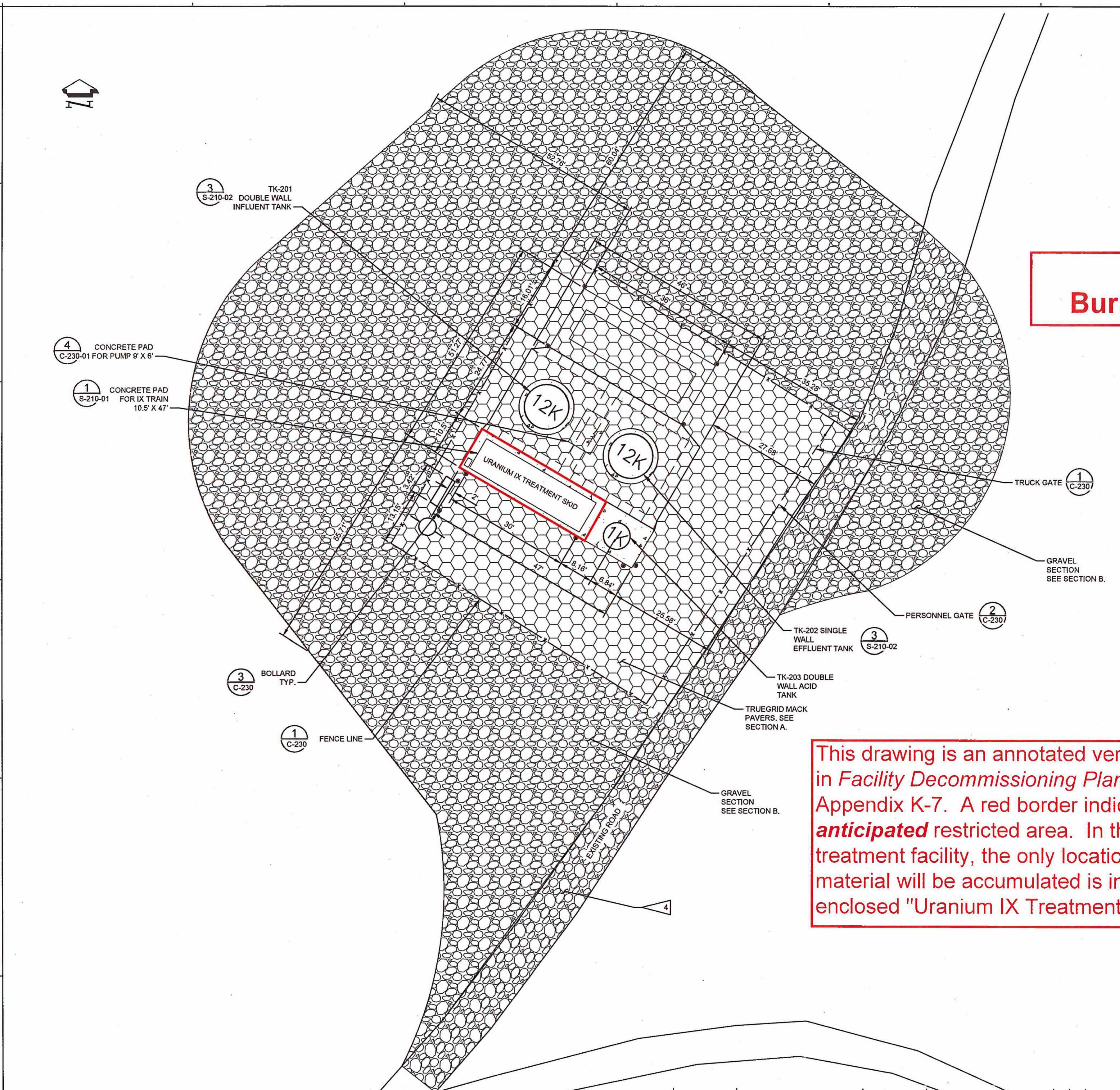
DRAWN BY S MOORE
CHECKER D KING

CIMARRON
PUMP AND TREAT SYSTEM PROJECT

WESTERN AREA TREATMENT FACILITY BUILDING POWER PLAN

SIZE F | DWG NO K-EPM-DWG-E-115 | REV B
SCALE SHOWN | SHEET 1 of 3

DWG NO | TITLE | DRAWING TRACEABILITY LIST | REF NUMBER | TITLE | REFERENCES | NEXT USED ON | REVISIONS



SITE PLAN

SCALE: 1" = 10'-0"

0 10 20 Feet

GENERAL NOTES:

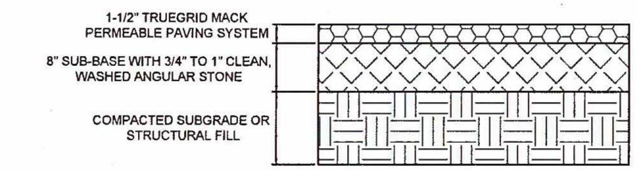
1. SEE K-EPM-DWG-C-201 FOR CIVIL NOTES.
2. SEE K-EPM-DWG-C-220 FOR SITE GRADING PLAN AND UTILITY PLAN.
3. SEE K-EPM-DWG-S-201 FOR STRUCTURAL NOTES.

4 ROAD FINISH IS TO MATCH NEW SURFACE CONDITIONS, SEE SECTION B FOR DETAILS.

TANK INFORMATION NOTES:

1. 12,000 GALLON TANK DIMENSIONS:
   DIAMETER = 10'-5" (126")
   HEIGHT = 18'-0" (216")
2. 1,100 GALLON TANK DIMENSIONS:
   DIAMETER = 6'-4" (76")
   HEIGHT = 8'-9" (105")

Figure 8-3
Burial Area #1 Treatment Skid

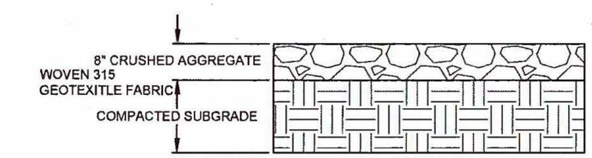


TRUEGRID MACK NOTES:

1. FOLLOW MANUFACTURER'S RECOMMENDATIONS FOR INSTALLATION OF TRUEGRID PRODUCT
2. USE $\frac{3}{8}$" TO $\frac{3}{4}$" AGGREGATE FOR GRAVEL FILL-IN OF GRID.

TRUEGRID MACK PAVING SECTION A

SCALE: NONE



NOTES:

1. FOLLOW MANUFACTURER'S RECOMMENDATIONS FOR INSTALLATION OF GEI WORKS WOVEN 315 OR EQUIVALENT GEOFABRIC PRODUCT.

GRAVEL SECTION B

SCALE: NONE

This drawing is an annotated version of a drawing in *Facility Decommissioning Plan - Rev 1 - Appendix K-7*. A red border indicates the ***anticipated*** restricted area. In the Burial Area #1 treatment facility, the only location where licensed material will be accumulated is in the fully enclosed "Uranium IX Treatment Skid".

IN-PROCESS
60% DESIGN PACKAGE UPDATE
NOT APPROVED FOR CONSTRUCTION

VEOLIA NUCLEAR SOLUTIONS
1355 Columbia Park Trail
Richland, WA 99352
www.nuclearsolutions.veolia.com

CIMARRON
PUMP AND TREAT SYSTEM PROJECT

BURIAL AREA #1
SITE PLAN

DRAWN BY D NELSON · ENGINEER D NELSON · PROJECT MANAGER J LUEY

SIZE F · DWG NO K-EPM-DWG-C-210 · REV B · SCALE SHOWN · SHEET 1 OF 1

DWG NO | TITLE — DRAWING TRACEABILITY LIST · REF NUMBER | TITLE — REFERENCES · NEXT USED ON · REVISIONS

10.0 RADIATION PROTECTION SURVEYS

10.1 General Requirements

Survey information is used to:

- assist in the development of Activity Plans (AP),
- inform individuals of the radiological conditions/hazards in the area,
- evaluate the need for area postings,
- identify needed personnel protective equipment,
- ensure personnel exposures to radiation and radioactive materials are maintained ALARA,
- determine the decommissioning status of material, equipment, and/or environmental media, and
- determine compliance with regulatory and/or license criteria.

Radiation and contamination surveys, air sampling, and sample collection will be performed as appropriate to assess radiological conditions and to establish specific radiological controls for work to be performed. Radiation protection surveys that are required by the license shall be conducted in accordance with specified requirements.

Two types of dose rates measurements may be used. Contact dose rates are used to locate and identify radiation levels detected and are measured within 1 cm (0.5 in) from the surface being surveyed. General area dose rates are used to identify radiation levels detected at approximately 30 cm (1 ft) from the surface being surveyed.

Surveys for removable and direct contamination are performed to detect and/or quantify radioactive contaminants. Removable contamination surveys should be performed when necessary to ensure that radioactive contamination has not inadvertently spread.

U.S. NRC Regulatory Guide 8.25, "Air Sampling in the Workplace" provides an acceptable method for meeting certain survey and dose assessment requirements of 10 CFR 20. Air samples shall be collected whenever the airborne radioactivity levels are expected to exceed ~~10 percent~~ 1 percent of the Derived Air Concentration (DAC) as listed in Appendix B, Table 1 "Occupational" of 10 CFR 20.

real-time radiation monitors. Dosimeters are posted at the Cimarron Site to confirm that no occupational worker is likely to receive 100 mrem DDE in a year.

10.6 Air Monitoring

1 percent

Air monitoring is required whenever airborne radioactivity levels are expected to exceed ~~10 percent~~ of the Derived Air Concentration (DAC) as listed in Appendix B, Table 1 "Occupational Values" of 10 CFR 20. Considering the types of work activities described in this Decommissioning Plan, airborne suspension of licensed radioactive material is not anticipated to generate airborne radioactivity approaching ~~10%~~ 1 % of a DAC. However, the Decommissioning Plan requires that General Area (GA) air sampling, using either low or high volume portable air samplers, will be performed throughout the resin unloading and packaging process for at least the first three resin exchanges. Following analysis of the air sample results from each of these resin exchanges, the RSO will determine the need and frequency of additional air sampling. ~~Selection of air samplers is based on the following criteria:~~

See insert to Section 10.6.

10.6.1.1 GA air sampling will be accomplished by using portable air samplers, as discussed, above. Sampling heads will be placed within the breathing zone to ensure that the air sample is representative of the air breathed by the individual worker.

1%

10.6.1.2 GA air samplers typically sample at a rate of approximately 3-25 liters per minute (lpm) for a low volume sampler to 70 cubic feet per minute (cfm) for a high volume sampler. Based on the nature of the low enriched uranium encountered, the detection capability of the air sampling equipment and associated radiological analysis (e.g., sample counting) will be used to determine the total volume of air needed to be collected to ensure that ~~10%~~ of the DAC. The enrichment of the uranium will be based on either the actual enrichment being collected on the resin or a conservative basis(i.e. 4%). This calculation will be documented in a site procedure or technical basis document. As the actual enrichment of recovered uranium in each area changes (i.e., WA or BA1), the ~~10%~~ DAC value may be recalculated Minimum collection times will be determined so adequate sensitivities are achieved for a given monitoring period.

1%

10.6.1.3 The need for air sampling will be prospectively determined based on the final process system design and potential for generation of airborne radioactivity. Due to the chemical and physical nature of the uranium-bearing media (e.g., water and moist ion exchange resin), minimal, if any airborne radioactivity is expected to be generated. Engineering and physical controls incorporated

INSERT FOR RPP SECTION 10.6:

> **Note:** A prospective evaluation of potential intake during groundwater processing operations was performed. The calculation supporting this evaluation is provided in Appendix A of the RPP. This calculation was based on 60% design of the groundwater treatment system and supported the decision that internal monitoring (e.g., bioassay) and respiratory programs were not needed at the Site. The evaluation also informed the development of the air sampling program described in Section 10.6 of the RPP. The supporting calculation will be reviewed at 90% design, updated, if necessary, and re-evaluated to determine if the RPP should be updated. In addition, periodically through groundwater processing, the supporting calculation will be reviewed to ensure it reflects operational experience and if changes to the RPP are necessary.

Selection of air samplers is based on the following criteria:

into the process equipment design will also be considered in determining the need for air monitoring.

10.6.1.4 The frequency of calibration of the flow meters on the air samplers will be based on manufacturers' recommendations (typically annually).

10.6.1.5 Action levels will be developed that will include specific action levels (i.e., specific projected or actual airborne radioactive material concentration levels) for assigning respiratory protection, collecting bioassay samples, and stopping work.

10.6.1.6 Air samples will be counted on-site using existing laboratory bench top scalers (e.g., Ludlum Model 3030 or similar equipment). Minimum detectable activities (MDAs) based on various sample count times will be calculated and used to determine the sample volume needed to detect less than ~~10%~~ 1% DAC for 4% enriched uranium. This information will be documented and used to determine the minimum sampling time for lapel air samplers.

10.7 Survey Training and Documentation

Surveys shall be performed by personnel who have been trained commensurate with the type of surveys to be performed. Training will address the following, as applicable:

- Appropriate instrumentation to be used,
- Operational and response checks for survey instrumentation,
- Survey methods, recording of data,
- Calculations, data evaluation, and
- Action levels.

Radiation and contamination surveys performed for compliance purposes, or to demonstrate that decommissioning criteria have been met, shall be documented and maintained in accordance with 10 CFR 20, Subpart L and the Quality Assurance Program Plan.

activities that could potentially expose workers to airborne radioactive material have been evaluated to determine the potential intakes during groundwater treatment and spent resin processing. The evaluation employed the methods discussed in Regulatory Guide 8.25, Rev. 1, "Air Sampling in the Workplace" and NUREG-1400, "Air Sampling in the Workplace." If processes or operations change, then a re-evaluation of potential intakes shall be performed to determine the potential intake that could result from these changes. If the potential intake determined from this evaluation is 2% ALI or greater, then the RSO will consider the need to implement a respiratory protection program as discussed in this section.

If a respiratory protection program is determined to be necessary, the program will be based on guidance provided in Regulatory Guide 8.15, Rev. 1, "Acceptable Programs for Respiratory Protection," and NUREG/CR-0041, Rev. 1, "Manual of Respiratory Protection Against Airborne Radioactive Material."

14.0 RESPIRATORY PROTECTION

14.1 Section Overview

Respiratory protection measures shall be employed when necessary to protect workers from airborne hazards. Groundwater treatment results in the generation of moist treatment media with little potential to generate airborne radioactivity. However, as future conditions change and the RSO or designee determines, through review of field conditions or anticipated work functions, that respiratory protection is required, procedures and controls will be instituted in accordance with the requirements found in 10 CFR 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas" for radiological hazards and the Code of Federal Regulations Title 29 Part 1910.134 for non-radiological hazards. Section 14.2 provides specific requirements for the respiratory protection program, if needed.

14.2 Respiratory Protection Program

Respiratory protection will be required if work activities could potentially expose workers to 40 or more derived air concentration (DAC)-hours in a week. Respiratory protection will also be required for any areas where airborne radioactive material concentrations are expected to exceed 1 DAC. If either of these trigger levels are encountered, a respiratory protection procedure or procedures) will be established to include:

- Process controls, engineering controls or procedures to control concentrations of radioactive material in air.
- Evaluations performed when it is not practical to apply engineering controls or procedures.
- Considerations used to demonstrate respiratory protection equipment is required.
- Required medical screening and respirator fit testing.
- Use, maintenance, and storage of respiratory protection devices.
- Respiratory protection training program.
- Selection of respiratory protection equipment.

APPENDIX A TO
CIMARRON RADIATION PROTECTION PLAN
CALCULATION OF POTENTIAL INTAKE

ENERCON Excellence · Every project. Every day.	CALCULATION COVER SHEET	CALC NO.	EPM028-CALC-001
		REV.	0
		PAGE NO.	1 of 10

Title:	**Potential Intake Calculation**	**Client:**	**Cimarron Environmental Response Trust**
		Project Identifier:	**EPM028**

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ____________	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** ____________	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Jay Maisler J Maisler	**Date: 3/26/2019**
Design Reviewer: A. Joseph Nardi A Joseph Nardi	**Date:** 5/6/19
Approver: Gerry Williams Gerald Williams Digitally signed by Gerald Williams DN: cn=Gerald Williams, o=ENERCON Services Inc., ou, email=gwilliams@enercon.com, c=US Date: 2019.05.07 08:20:47 -04'00'	**Date:**

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. EPM028-CALC-001
		REV. 0
		PAGE NO. 2 of 10

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	3/26/2019	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
			Attachment A	2	0

CALCULATION TABLE OF CONTENTS

ENERCON Excellence—Every project. Every day.	TABLE OF CONTENTS	CALC NO.	EPM028-CALC-001
		REV.	0

1.0 Purpose and Scope

The purpose of this calculation is to estimate potential intakes from groundwater processing and resin handling at the Cimarron Site. The methodology for potential intake and need for air sampling is based on the methodology provided in NUREG-1400 (Ref. 3.1).

2.0 Summary of Results and Conclusions

The highest anticipated potential intake from handling spent resin is less than 0.2% of the annual limit on intake (ALI) for uranium.

3.0 References

3.1 NUREG-1400, Air Sampling in the Workplace, September 1993.
3.2 49 CFR 173.434, Activity-mass relationships for uranium and natural thorium.
3.3 10 CFR 20, Appendix B, Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage
3.4 Cimarron Radiation Protection Plan, draft Rev. 4
3.5 BA1 Isotopic Data for Enercon.xlsx, Isotopic abundance of Uranium source provided as an Excel spreadsheet
3.6 Cimarron Facility Decommissioning Plan, Rev. 1

4.0 Assumptions

4.1 The specific activity for uranium:

4.1.1 5% enrichment – 2.7 E-06 Ci/g (Ref. 3.2)
4.1.2 1.5% enrichment – 1.00 E-06 Ci/g (Ref. 3.2)

4.2 The maximum groundwater uranium concentration is assumed to be 5000 pCi/L. This is conservative based on the highest reported groundwater concentration found in BA1 of less than 3000 µg/L. (Ref. 3.6, Figure 3-4)

4.3 The maximum mass of U-235 assumed to be present in the spent resin being processed for disposal is 500 grams. This is higher than the maximum mass estimated in the treatability study. (Ref. 3.6, section 8.3.2)

4.4 The uranium in groundwater is assumed to be soluble.

4.5 Determining "potential intake," based on the NUREG-1400 methodology, requires various assumptions:

4.5.1 Total activity processed is based on twelve spent resin bed vessels processed per year.
4.5.2 "Release Fraction," R, is based on "non-volatile powder" – 0.01.
4.5.3 "Confinement Factor," C, is equivalent to a well-vented hood – 0.1. This is based on equipment designed to confine resin within the process system.
4.5.4 "Dispersibility," D, is based on the contaminant being moist resin – 0.1

4.6 The ALI and DAC used for inhalation are for U-238, class Y. U-234, U-235, and U-238 have the same value for these ALIs and DACs, therefore isotopic distribution is irrelevant to the dosimeteric calculation.

5.0 Design Inputs

5.1 Annual Limit on Intake (ALI) for Inhalation
The following properties are taken from Reference 3.3.

4.00 E-02 µCi

5.2 Derived Air Concentration (DAC) for Inhalation
The following source characteristics are taken from Reference 3.3.

2.00 E-11 µCi/mL

5.3 ALI for Oral Ingestion
The following material definitions are taken from Reference 3.3.

1.00 E01 µCi (class D, bone surface)
2.00 E01 µCi (class D)

5.4 Resin Vessel Loading
The first week's loading of the resin vessel is 3,706 grams of uranium. After the first full loading, the vessel contains 24,055 grams of uranium. These values are taken from Reference 3.5.

6.0 Methodology

6.1 Oral Intake Consumption
The amount of untreated groundwater that would need to be consumed for an individual to have an intake of 2% ALI and 1 ALI was calculated based on Assumption 4.2.

6.2 Spent Resin Loading
The total amount of uranium that could be loaded onto a spent resin bed was calculated for 1.5% and 5% enriched uranium. These calculations provided for conservatism in the dose assessment performed.

6.3 NUREG-1400 Methodology
NUREG-1400 provides calculational methods to support decisions related to the need to perform air sampling at a facility and determine the potential intake for workers.

The first calculation involves determining if the amount of unsealed radioactive materials handled in a year would indicate the need for performing air sampling. Following this methodology, the amount uranium on a spent resin bed was determined. Then, as discussed in Assumption 4.4.1, the total amount of uranium handled in a year was determined. Other assumptions (4.4.2, 4.4.3, 4.4.4) were used to calculate "potential intake" from inhalation.

6.4 Chemical Intake
Intake of soluble uranium is limited to 10 mg per week as required by 10 CFR 20.1201(e). Using the "potential intake" from inhalation, the mass that an individual could intake in a year was calculated. Because the limit is based on a weekly limit, consumption of contaminated water would be more limiting than inhalation. Based on Assumption 4.2, the amount of contaminated water that would need to be consumed during a week to ingest 10 mg of soluble uranium was calculated.

7.0 Calculation

7.1 Oral Intake Consumption

The oral intake consumption is found using the Equation 7.1.

$$I_o = \frac{ALI}{C_{max}} \times 1E06$$

Equation 7.1

Where I_o, in liters, is equal to annual limit on intake (ALI) μCi for U-238 divided by the maximum activity concentration (pCi/L) in groundwater multiplied by a 10^6 pCi/μCi.

The results from this calculation indicate that drinking 40 liters of contaminated groundwater at maximum activity would result in an individual intake of 2% ALI; 2000 L would need to be consumed to have an intake of 1 ALI. Consuming groundwater at the site is prohibited.

7.2 Spent Resin Loading

The mass of total uranium in a fully spent resin bed is found using Equation 7.2.

$$M = \frac{500\ g}{E}$$

Equation 7.2

Where the total mass of uranium in spent resin vessel, M, equals the mass of U-235 on the spent resin bed (500 grams) divided by the enrichment.

Based on 1.5% enrichment, the uranium mass in one spent resin bed would be 33.3 kg.

7.3 NUREG-1400 Methodology

The total activity in spent resin processed in a year is based on four resin bed exchanges per year. The total activity handled during a year is calculated with Equation 7.3.

$$Q = SA_{enrich} \times M \times 4$$

Equation 7.3

Where Q, the total activity handled in a year (Ci) is equal to the specific activity based on enrichment, SA_{enrich}, Ci/g divided by the total mass of uranium on a fully spent resin bed from Equation 7.2. The activity Q is compared to the ALI for uranium to determine if it exceeds 10^4 times the ALI. If it does, then air sampling should be considered.

The total activity assuming 1.5% enrichment would exceed 10^4 times the ALI. Air sampling should be considered and is addressed in the RPP (Ref. 3.4)

Potential intake from inhalation is determined from Equation 7.4 (Ref. 3.1, Equation 1.2).

$$I_p = Q \times R \times C \times D \times 10^{-6}$$

Equation 7.4

Where I_p is the potential intake from inhalation in µCi/mL, Q is the total activity from Equation 7.3 converted to µCi, R is the release fraction (0.01 for non-volatile powders), C is the confinement factor (0.1 for well-ventilated hood), and D is dispersibility (0.1 for moist resin). 10^{-6} is an additional factor provided in Ref. 3.1

The potential intake from inhalation for 5% enrichment was more limiting than 1.5% enrichment. The results for 5% enriched uranium was 2.59 E-05 µCi in a year, which is 0.2% ALI. At 1.5% enrichment, the potential intake was calculated to be 0.1% ALI.

7.4 Chemical Intake

The amount of groundwater that would need to be consumed to have an intake of 10 mg of soluble uranium was determined using Equation 7.5.

$$V = \frac{10\ mg}{3000\ \mu g/L} \times 10^3$$

Equation 7.4

Where V is the volume consumed in liters. 10 mg is the weekly soluble uranium intake limit (10 CFR 20.1201(e)). 3000 µg/L is the maximum mass concentration of uranium in groundwater at BA1.

The calculation resulted in a weekly intake exceeding 10 mg soluble uranium if an individual consumed 3.33 L of contaminated water. Consumption of groundwater at the Cimarron site is prohibited.

Chemical intake from inhalation was also calculated and compared to the ALI for inhalation.

$$I_w = \frac{10\ mg}{ALI} \times 1E06$$

Equation 7.5

Where I_w is the weekly intake in µCi.

An intake of 10 mg soluble uranium by inhalation would involve an individual breathing in 250 times the ALI at 1.5% enrichment.

8.0 Computer Software

A Microsoft Excel spreadsheet was used to perform calculations discussed in this calculation.

Attachment A – Potential Intake Calculation

EPM028-CALC-001									
Assumptions					**More realistic calculation**				
Specific Activity (5% enrich.)	2.70E-06	**Ci/g**			SA (1.5%)	1.00E-06	**Ci/g**		
Max GW concentration	5000	**pCi/L**	5.00E-06	**µCi/mL**					
Oral Intake ALI	1.00E+01	**µCi**							
Consumption for 2% ALI	40.00	**L**							
Consumption for 1 ALI	2000.00	**L**							
Spent resin loading									
Site limit for U-235	1200	**g**			Max U235	500	**g**		
Total U assume 5%	2.40E+04	**g**			U (1.5%)	3.33E+04	**g**		
NUREG-1400 Methodology									
Total activity per resin bed	6.48E-02	**Ci**	6.48E+04	**µCi**		3.33E-02	**Ci**	3.33E+04	**µCi**
Total activity (Q) (12 bed/y)	7.78E-01	**Ci**	7.78E+05	**µCi**		4.00E-01	**Ci**	4.00E+05	**µCi**
Is Q > 1E+04 ALI?	7.78E+04	**YES**				4.00E+04	**YES**		

Potential Intake (I_P)	I_P = Q x 1E-06 x R x C x D								
Total activity (Q) (12 bed/y)	7.78E+05	**µCi**				4.00E+05	**µCi**		
Release Fraction (R)	0.01	Nonvolatile Powder				0.01	Nonvolatile Powder		
Confinement Factor (C)	0.1	Well-ventilated hood	Glovebox would be 0.01			0.1	Well-ventilated hood		
Dispersibility (D)	0.1	Moist resin				0.1	Moist resin		
Potential Intake (I_P)	7.78E-05	**µCi**				4.00E-05	**µCi**		
%ALI	0.19%					0.10%			
Chemical Intake									
Potential Intake (I_P)	7.78E-05	**µCi**				4.00E-05	**µCi**		
Mass intake per year	2.88E-05	**g**	2.88E-02	**mg**		4.00E-05	**g**	4.00E-02	**mg**
Drinking	3000	**µg/L**				3000	**µg/L**		
Limit per week	10	**mg**				10	**mg**		
Volume per week	3.33E+00	**L**				3.33E+00	**L**		
Activity/week	2.70E-05	**Ci**				1.00E-05	**Ci**		
	2.70E+01	**µCi**				1.00E+01	**µCi**		
%ALI	67500.00%					25000.00%			

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
2. Are the proper forms being used and are they the latest revision? Same format matching EPM017-CALC-001 was used for internal consistency	☐	☒	☐
3. Have the appropriate client review forms/checklists been completed? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation? No current ENERCON calculations exist that are similar to this calculation.	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☐	☒	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation? No ENERCON design inputs or outputs are affected by this calculation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis? This calculation applies to a remediation site. No work performed in this calculation is applicable to a licensing basis.	☐	☐	☒
17. If so, is this documented in the calculation?	☐	☐	☒
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☐	☒
19. If so, is this documented in the calculation?	☐	☐	☒
20. Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☐	☒
21. If so, is this documented in the calculation?	☐	☐	☒
22. Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS			
23. Are design inputs clearly identified?	☒	☐	☐
24. Are design inputs retrievable or have they been added as attachments?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
25. If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable? MS Excel spreadsheet was used to perform the calculation. All equations are provided in the calculation.	☐	☐	☒
26. Are design inputs clearly distinguished from assumptions?	☒	☐	☐
DESIGN INPUTS (Continued)			
27. Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☐	☒	☐
28. Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29. Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☐	☐	☒
30. If applicable, do design inputs adequately address actual plant conditions?	☐	☐	☒
31. Are input values reasonable and correctly applied?	☒	☐	☐
32. Are design input sources approved? The Cimarron design is currently at 60% completion.	☐	☒	☐
33. Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34. Were all applicable plant operating modes considered?	☐	☐	☒
ASSUMPTIONS			
35. Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36. Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37. Are any engineering judgments used?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
38. Are engineering judgments clearly identified as such? Engineering judgement applied to factors used in potential intake calculation using NUREG-1400 methodology.	☒	☐	☐
39. If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☒	☐	☐
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☐	☒
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use? Equations applied in this evaluation are in common use in the industry.	☐	☐	☒
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input? Results generated by calculations performed in this evaluation are not significantly affected by minor perturbations of variables.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☐	☒	☐
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒
SOFTWARE/COMPUTER CODES (Continued)			
52. Are the codes properly identified along with source vendor, organization, and revision level?	☐	☐	☒
53. Is the computer code applicable for the analysis being performed?	☐	☐	☒
54. If applicable, does the computer model adequately consider actual plant conditions?	☐	☐	☒
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☐	☐	☒
56. Is the computer output clearly identified?	☐	☐	☒
57. Does the computer output clearly identify the appropriate units? The output units are not identified in the output document. Tallies have been modified through multipliers and dose response functions. This process has been adequately documented within this calculation.	☐	☐	☒
58. Are the computer outputs reasonable when compared to the inputs and what was expected? Only basic functions and operations in Microsoft Excel 2013 were applied in this calculation.	☒	☐	☐
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☐	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified? No acceptance criteria required for this evaluation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☐	☐	☒
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☐	☐	☒
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☐	☐	☒
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations? No ENERCON calculations are affected by this evaluation.	☐	☐	☒
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☒	☐	☐

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator:

JMaisler

3/26/2019

Jay Maisler

Print Name and Sign

Date

APPENDIX B TO
CIMARRON RADIATION PROTECTION PLAN
CALCULATION OF POTENTIAL DOSE RATE
NEAR URANIUM TREATMENT TRAIN

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC NO.	EPM017-CALC-001
		REV.	0
		PAGE NO.	1 of 15

Title:	Dose Rate Near Uranium Treatment Train	Client:	Cimarron Environmental Response Trust
		Project Identifier:	EPM017

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ____________________	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** ____________________	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation	☐	**Final Calculation**	☒
Safety-Related	☐	**Non-Safety-Related**	☒

(Print Name and Sign)

Originator: Caleb Trainor	Caleb T	**Date: 12/21/2015**
Design Reviewer: John Hawkinson	John Hawkinson	**Date: 12/23/2015**
Approver: Jay Maisler, CHP	J Maisler	**Date: 12/23/2015**

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	12/21/2015	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.

CALCULATION TABLE OF CONTENTS

ENERCON Excellence—Every project. Every day.	TABLE OF CONTENTS	CALC NO.	EPM017-CALC-001
		REV.	0

1.0 Purpose and Scope

The purpose of this calculation is to determine the radiation level surrounding a resin vessel array used to extract uranium contamination from groundwater at the Cimarron Site, hereby referred to as a uranium treatment train. The results of this calculation represent the highest anticipated dose rate an individual would be subject to while in the close vicinity of a uranium treatment train.

2.0 Summary of Results and Conclusions

The highest anticipated dose rate in the vicinity of a uranium treatment train is calculated to be 0.024 mrem/hr.

3.0 References

3.1 MCNP6-1.0, Revision 00, MCNP6 Version 1.0 Acceptance Report, November 2014
3.2 LA-CP-13-000634, MCNP6 User's Manual, Version 1.0, May 2013
3.3 MicroShield 6.20
3.4 DTS-WPS-M-3100, AVANTech Drawing, '48" Process Vessel', Rev B, January 10, 2000
3.5 Vessel Use and U235 Accum Calc_DRAFT_16Oct15.xlsx, Vessel loading analysis provided as an Excel spreadsheet
3.6 PNNL-15870, Rev. 1, Compendium of Material Composition Data for Radiation Transport Modeling, March 4, 2011
3.7 Email correspondence with Kurion representative (attached)
3.8 BA1 Isotopic Data for Enercon.xlsx, Isotopic abundance of Uranium source provided as an Excel spreadsheet
3.9 NUREG/BR-0150, Vol. 1, Rev. 4, RTM-96 Response Technical Manual, March 1996
3.10 Introduction to Health Physics, 4th ed., Herman Cember and Thomas E. Johnson, McGraw-Hill, 2009.

4.0 Assumptions

4.1 The resin vessel is approximated as a perfect right circular cylinder, maintaining the actual volume and clad thickness of the vessel. This is a minor simplification made for modeling purposes and is insignificant to the final dose rate.

4.2 The resin vessel is modeled as stainless steel 304. At the time of this calculation, the vessel was known to be composed of stainless steel, but the specific alloy was unknown. Stainless steel 304 was chosen as it is a popular alloy for nuclear vessels. The specific alloy of stainless steel has insignificant impact on shielding.

4.3 The mass of uranium assumed to be present in the resin vessels is based off of a draft calculation provided for this study. The calculation is performed in Excel spreadsheet "Vessel Use and U235 Accum Calc_DRAFT_16Oct15.xlsx" (Reference 3.5). From this calculation, the largest expected mass accumulation of uranium is 24,055 g of uranium at Burial Area 1 (BA1) after the first full loading. The theoretical capacity of the resin used for the calculation is 29,589.01 grams.

The vessel loading calculations use an enrichment of 2% by weight. This value was chosen as a conservative bound for the actual 1.66% enrichment in order to remain conservative with respect to materials controls. The resin loading is not dependent on enrichment and the input values used in this calculation are not affected by this assumption.

4.4 The lead vessel is modeled assuming it is completely full to its theoretical capacity, ready to be removed and replaced with the lag vessel. The lag vessel is assumed to accumulate one week's worth of uranium, due to a one week period between sampling intervals. The accumulation in the first week of loading in the lead vessel is used to define one week's worth of accumulation. This is conservative as the uranium loading rate decreases with time; as the concentration of contaminants in the fluid passed through the vessels decreases, the resin removes a lower volume of uranium. The polishing vessel is assumed to provide negligible dose rate contribution due to the significant distance and low accumulation of uranium.

4.5 The uranium is assumed to be evenly distributed through the resin vessel.

4.6 The DOWEX-1 resin is approximated as polystyrene. For shielding purposes, this media is roughly equivalent. The mass of resin used is provided in Reference 3.5. The resin is assumed to fill the vessel and is homogenized, while maintaining mass, for modeling purposes.

4.7 The shielding provided by the vessel clad and resin are modeled. The shielding which would be provided by the water and additional piping inside the vessel has been omitted. Miscellaneous piping and structures outside the vessels are also omitted. The lack of additional shielding provides a conservative estimate, which results in a higher estimation of dose rate.

4.8 All alpha and beta particles are shielded by the 3/8 inch stainless steel walls of the resin vessel and do not contribute to the final dose rate.

4.9 The spontaneous fission rate for the uranium isotopes is sufficiently low (3.5 n/s per kg U-234, 0.31 n/s per kg U-235 and 7.0 n/s per kg U-238) that the neutron dose rate can be disregarded.

5.0 Design Inputs

5.1 Resin Vessel Properties

The following properties are taken from Reference 3.4.

Vessel Diameter=48 in
Volume=54.5 ft^3

The following properties are taken from Reference 3.7.

Vessel Clad=3/8 in
Vessel Spacing=48 in (96 in on center)

5.2 Source Characteristics

The following source characteristics are taken from Reference 3.5

Media Capacity=39.52 g U/kg
Media Mass used=748.7 kg

The following isotopic abundances by mass percent are taken from Reference 3.8.

U_{234}= 0.01
U_{235}= 1.66
U_{238}= 98.33

The following specific activities are taken from Table E-4 of Reference 3.9.

U_{234}= 6.19 E+03 µCi/g
U_{235}= 2.14 E+00 µCi/g
U_{238}= 3.33 E-01 µCi/g

5.3 Material Definitions

The following material definitions are taken from Reference 3.6.

Table 5.1: Composition of Polystyrene (Density homogenized to 0.485139 g/cc)

Nuclide	Weight Fraction
Hydrogen	0.077421
Carbon	0.922579

Table 5.2: Composition of Stainless Steel 304 (Density of 8.0 g/cc)

Nuclide	Weight Fraction
Carbon	0.000400
Silicon	0.005000
Phosphorus	0.000230
Sulphur	0.000150
Chromium	0.190000
Manganese	0.010000
Iron	0.701730
Nickel	0.092500

5.4 Resin Vessel Loading

The first week's loading of the resin vessel is 3,706 grams of uranium. After the first full loading, the vessel contains 24,055 grams of uranium. These values are taken from Reference 3.5.

6.0 Methodology

6.1 Source Term

The activity by isotope is determined using the mass abundance of each isotope from Reference 3.8 and applying the specific activities from Reference 3.9. These activities are entered into MicroShield 6.20 (Reference 3.3) to produce a list of gamma emission energies and activity in photons per second. These energies correspond to the energy bins defined in the MCNP6 (Reference 3.2) source term distribution. The activity in photons per second correspond to the energy distribution probabilities defined in the MCNP6 source term.

6.2 Shielding Evaluation

The MCNP6 code is used to model the resin vessel and cylindrical source term inside the vessel. The interior of the vessel is modeled as homogenized polystyrene. Outside of the vessel is comprised of dry air, characteristic of near sea level. Two point detectors used, one located mid height of the vessel, 12 inches (30.48 cm) from the outer surface of the vessel and another located 96 inches

(283.84 cm) offset laterally. The first detector represents the dose rate contribution from the modeled vessel (the lead vessel), the second detector represents the dose rate contribution from the lag vessel. Figures 6.1 and 6.2 are representative of the MCNP6 input.

Figure 6.1: Z-plane of Modeled Geometry

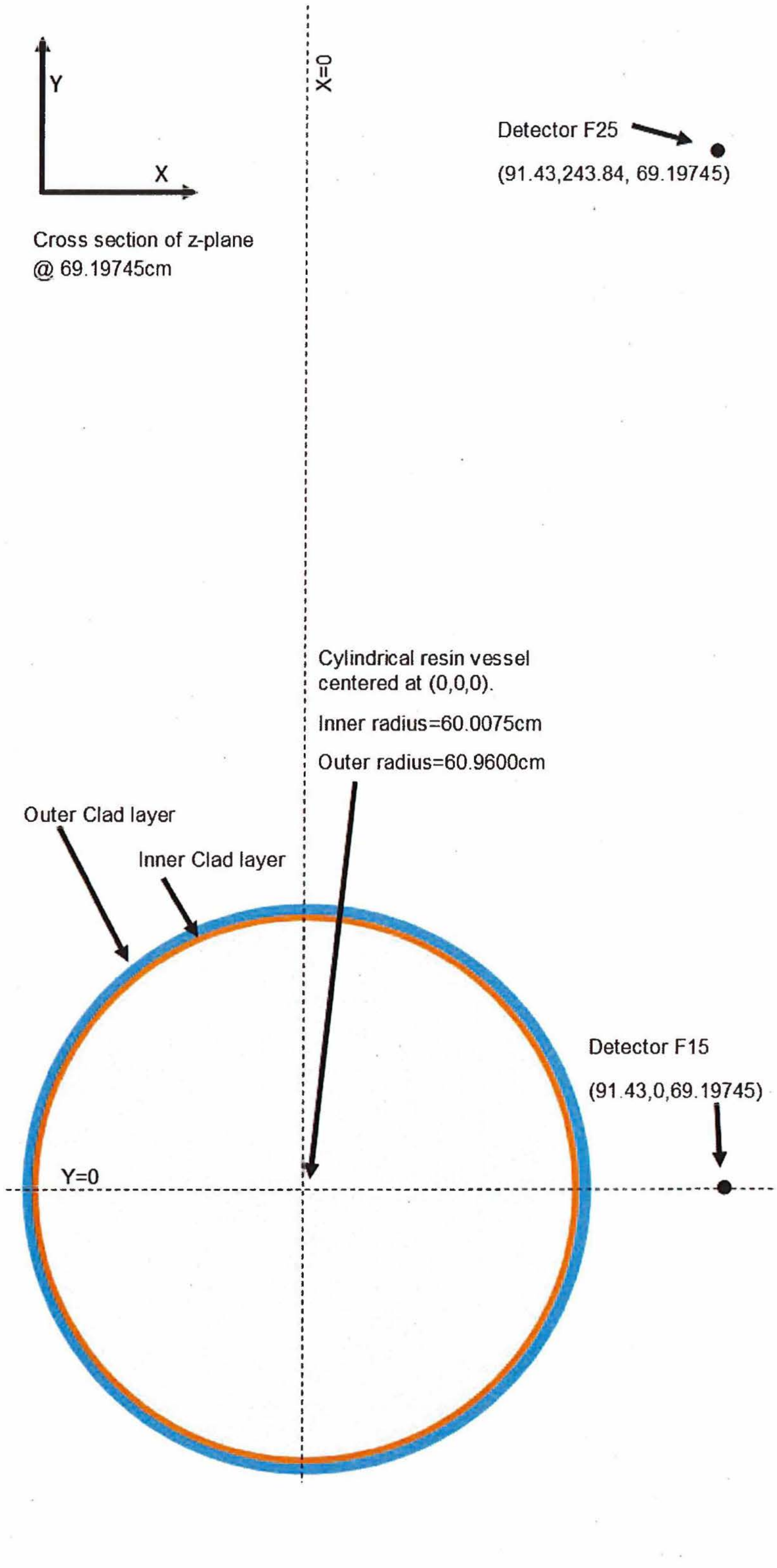
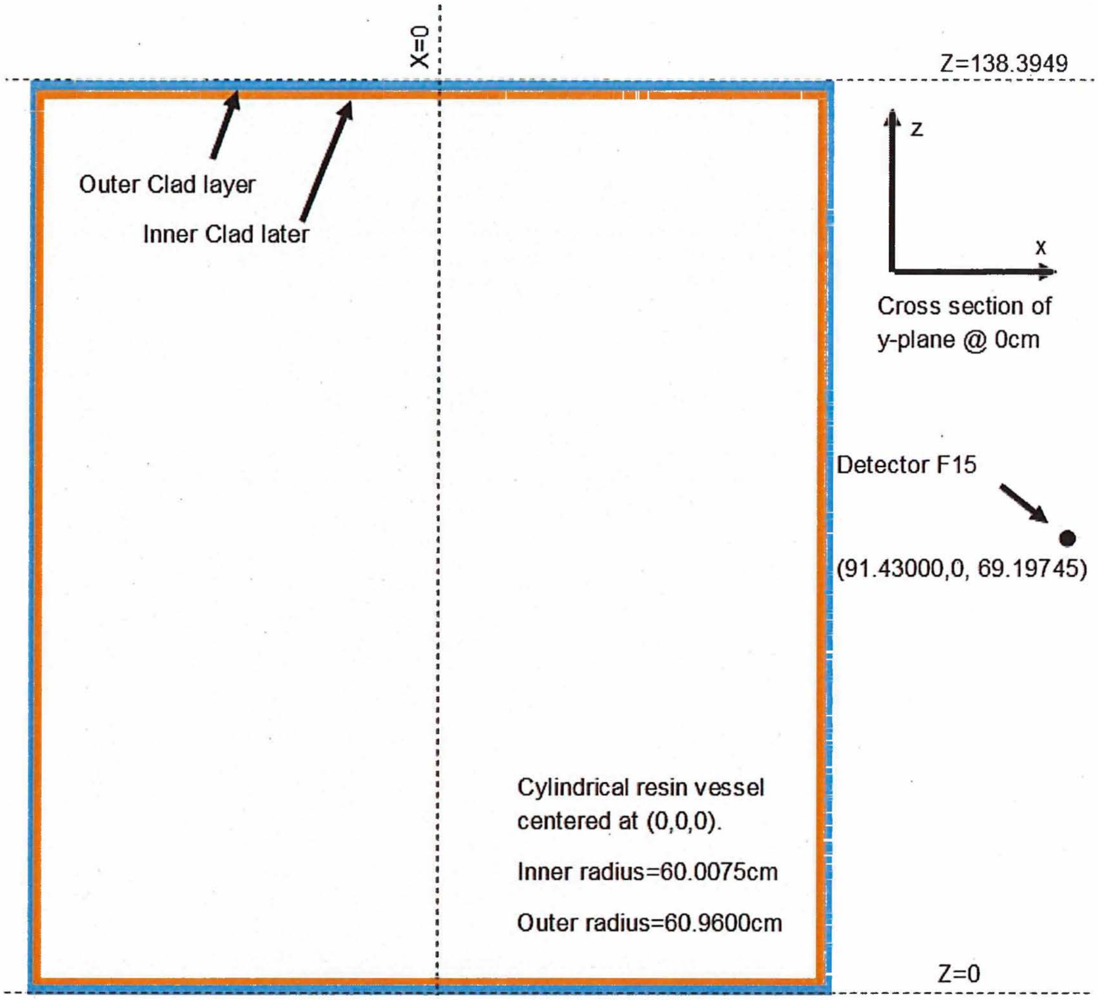


Figure 6.2: Y-Plane of Modeled Geometry



7.0 Calculation

7.1 Source Development

The activity of each isotope is found using the Equation 7.1.

$$A_i = U_i(g) * SA_i\left(\frac{\mu Ci}{g}\right)$$

Equation 7.1

Where A_i is equal to activity of each isotope, in μCi, U_i is the mass of uranium isotope i, and SA_i is equal to the specific activity of uranium isotope i.

The mass of each isotope in a full resin vessel is found using Equation 7.2.

$$U_i(g) = Media\ capacity\left(\frac{U(g)}{kg}\right) * Media\ mass\ (kg) * w_i$$

Equation 7.2

Where w_i represents the isotopes abundance by weight percent. Solving for the three isotopes;

$$U_{234}(g) = 39.52\left(\frac{U(g)}{kg}\right) * 748.7\ (kg) * 0.0001 = 2.958\ g\ U_{234}$$

$$U_{235}(g) = 39.52\left(\frac{U(g)}{kg}\right) * 748.7\ (kg) * 0.0166 = 491.177\ g\ U_{235}$$

$$U_{238}(g) = 39.52\left(\frac{U(g)}{kg}\right) * 748.7\ (kg) * 0.9833 = 29094.873\ g\ U_{238}$$

Using these values, and the specific activities defined in Design Input 5.2, Equation 7.1 is solved for the three isotopes;

$$A_{U_{234}} = 2.958\ (g) * 6190\ \left(\frac{\mu Ci}{g}\right) = 1.83 * 10^4 \mu Ci$$

$$A_{U_{235}} = 491.177\ (g) * 2.140\ \left(\frac{\mu Ci}{g}\right) = 1.05 * 10^3 \mu Ci$$

$$A_{U_{238}} = 29094.873(g) * 0.333\ \left(\frac{\mu Ci}{g}\right) = 9.69 * 10^3 \mu Ci$$

With the activities determined, MicroShield 6.20 is used to define gamma emission energies and emission frequencies. The MicroShield output is listed in Table 7.1.

Table 7.1: Gamma Emissions from Uranium source

Energy (MeV)	Activity (Photons/sec)
0.013	7.112E+07
0.013	3.166E+07
0.013	1.201E+07
0.0532	7.990E+05
0.0664	3.478E+05
0.0727	4.273E+04
0.09	1.060E+06
0.0933	1.732E+06
0.105	8.014E+05
0.1091	5.827E+05
0.12	5.827E+04
0.1214	2.712E+05
0.1408	8.547E+04
0.1438	4.079E+06
0.1633	1.826E+06
0.1827	1.554E+05
0.1857	2.098E+07
0.1904	3.572E+05
0.1949	2.292E+05
0.2021	3.885E+05
0.2053	1.826E+06
0.2214	3.885E+04

7.2 MCNP6 Input Development

7.2.1 Geometry Specifications

The only resin vessel modeled is the lead vessel. Since the lead vessel accumulates the most uranium, the area of interest is directly in front of the lead vessel. A detector is placed halfway up the vessel, 12 inches (30.48 cm) from the outer surface of the vessel. To model the dose rate contribution from the lag vessel, a second detector is placed at the same height, offset laterally by 96 inches (283.84 cm). The inner and outer clad layers represent the vessel clad, seperated into two layers for variance reduction purposes.

The geometry between the modeled vessel and the second detector is equivalent to the geometry between the lag vessel and the dose rate point of interest. The second detector tally will be scaled accordingly to represent the source term which would be present in the lag vessel.

The lead vessel is modeled using two concentric right circular cylinders with a separation of 3/8 inches around the top, sides, and bottom. The dimensions of the interior cylinder are calculated, preserving the defined vessel volume and diameter.

$$Vol = h_i * \pi {r_i}^2$$

Equation 7.4

Where;

$$Vol = 54.5\ ft^3$$

$$r_i = \frac{D - (2 * clad)}{2} = \frac{48\ in. - (2 * \frac{3}{8} in.)}{2} = 23.625\ in$$

Solving for h_i;

$$h_i = \frac{54.5\ ft^3 * 1728\ \frac{in^3}{ft^3}}{3.14159 * (23.625\ in)^2} = 53.7362\ in$$

The clad thickness is accounted for in the dimensions for the outer cylinder in the following equations.

$$h_o = h_i + 2 * Clad$$

$$= 53.7362\ in + 2 * {}^3/_8\ in = 54.4862\ in$$

$$r_o = r_i + Clad$$

$$= 23.625\ in + {}^3/_8\ in = 24\ in$$

For variance reduction purposes, a third right circular cylinder is included. This cylinder divides the vessel clad into two equal layers, shown as the "inner clad layer" and "outer clad layer" in Figures 6.1 and 6.2. The following excerpt from the MCNP6 input deck shows the cell and surface cards which define the geometry described above.

Figure 7.1: Cell and Surface Cards from Input Deck

```
c Cells
10 2 -0.485139 -100          imp:p=1                      $Resin (polystyrene) inside vessel
50 1 -8 -10 100              imp:p=2                      $SS 304 layer
11 1 -8 -101 10 100          imp:p=4                      $SS 304 vessel clad
30 0 -300 101                imp:p=8                      $Void
40 0 300                     imp:p=0                      $Boundary

c Surfaces
100 RCC 0 0 0.9525 0 0 136.4899 60.0075                   $Interior cylinder of resin vessel,
10  RCC 0 0 0.47625 0 0 136.96615 60.48375                $Layer used for variance reduction
101 RCC 0 0 0 0 0 138.3949 60.96                          $Outer cylinder of resin vessel
102 S 91.43 0 69.19745 20                                 $Detector 1 Void
103 S 91.43 243.84 69.19745 20                            $Detector 2 Void
300 RPP -65 100 -65 250 0 150                             $Bounding Area
```

7.2.2 Source Definition

The source is defined as a photon source, evenly distributed within a right circular cylinder equivalent to the interior cylinder of the resin vessel. The values in Table 7.1 are used to define the energy and probability of gamma particles in this area. The terms axs, pos, rad, and ext define the vector, base position, radius and height of the source term respectively.

Figure 7.2: Source Definition from Input Deck

```
c Source Definition
SDEF PAR=P axs=0 0 1 pos=0 0 0 rad=d1 ext=d2 erg=d3 cel=10      $Cylindrical source located inside inner cylinder
SI1 0 60.0075
SI2 0.9525 137.4424
SI3 L 0.13 .013 .013 .0532 .0664 .0727 .09 .0933 .105           $Discrete energy distribution based on gamma energies from MicroShield
     .1091 .12 .1214 .1408 .1438 .1633 .1827 .1857 .1904
     .1949 .2021 .2053 .2214
SP3  7.112e7 3.166e7 1.201e7 7.990e5 3.478e5 4.273e4 1.060e6     $Gamma emission frequency from MicroShield
     1.732e6 8.014e5 5.827e5 5.827e4 2.712e5 8.547e4 4.079e6
     1.826e6 1.554e5 2.098e7 3.572e5 2.292e5 3.885e5 1.826e6
     3.885e4
```

7.2.3 Tally Definitions

Two tallies are used to find the total dose rate. The first tally is located mid height and one foot away from the modeled lead resin vessel. The second tally is located in the equivalent position on the lag resin vessel (not modeled). Using these tallies, the total dose rate is equal to the summation of the first and second tally.

A tally multiplier of 1.504 E+08 is applied to the first tally. This value is equal to the summation of column 2 of Table 7.1 and scales the results to the source strength. A tally multiplier of 2.568 E+07 is applied to the second tally. This is approximately 15% of the source strength used for the first tally. This tally represents the dose rate contribution of the lag vessel, acting as the primary resin vessel for one week's time per Assumption 4.4. A ratio of 15%

is equivalent to the ratio of the first weeks vessel loading compared to a fully loaded vessel. The values from Design Input 5.4 are used to determine this ratio.

An MCNP6 dose response function is also applied to all tallies. This function sorts the tallies into energy bins, applies dose conversion factors from ANSI 6.1.1-1977, and sums all the bins. Due to the tally multiplier and dose function, the output of the tallies is in rem per hour. Figure 7.3 shows the tally definitions from the input deck.

Figure 7.3: Tally Definitions from Input Deck

```
c Tally Definition
F15:p 91.43 0 69.19745 10                    $Point Detector, mid height of vessel, one foot away
F25:p 91.43 243.84 69.19745 10               $Point Detector, mid height of vessel, one foot away, 96" offset
c                                            $equivalent to dose contribution from lag vessel
FM15 1.5045072E8                             $Tally multiplier equivalent to summation of all gamma emission frequencies
FM25 2.2567608e7                             $Source scaled to one week of buildup in lag vessel
c
c   ANSI 6.1.1-1977 Gamma Flux to Dose Conversion Factors, using US units
c   (rem/hr)/(photons/cm2-s)
df0  IU 1 IC 20                              $Seperates tally into energy bins, applies conversion factors and sums
```

7.3 Dose Rates

The estimated dose rates present near the uranium treatment train are presented in Table 7.2.

Table 7.2: Dose Rate

Tank Contribution	Equivalent Tally	Dose Rate (rem/hr)
Lead tank	F15	2.392 E-05
Lag tank	F25	4.974 E-07
	Total Dose Rate	2.442 E-05 (.02442 mrem/hr)

8.0 Computer Software

MCNP6 is used for shielding analysis in this calculation. It is verified and validated for use (Reference 3.1).

MicroShield 6.20 is used for calculating gamma emission energies and frequencies.

Attachment A

MCNP6 Input

```
Cimmarron Resin Bed Dose Estimation
c Cells
10 2 -0.485139 -100        imp:p=1                  $Resin (polystyrene) inside vessel
50 1 -8 -10 100            imp:p=2                  $SS 304 layer
11 1 -8 -101 10 100        imp:p=4                  $SS 304 vessel clad
30 0 -300 101              imp:p=8                  $Void
40 0 300                   imp:p=0                  $Boundary

c Surfaces
100 RCC 0 0 0.9525 0 0 136.4899 60.0075             $Interior cylinder of resin vessel, 3/8" (0.9525cm) off of floor to accomodate vessel clad
10  RCC 0 0 0.47625 0 0 136.96615 60.48375          $Layer used for variance reduction
101 RCC 0 0 0 0 0 138.3949 60.96                    $Outer cylinder of resin vessel
102 S 91.43 0 69.19745 20                           $Detector 1 Void
103 S 91.43 243.84 69.19745 20                      $Detector 2 Void
300 RPP -65 100 -65 250 0 150                       $Bounding Area

c Data
Mode P
nps 10e6
c
c Material Definitions
M1   6000 -0.000400 14000 -0.005000 15000 -0.000230  $SS 304
     16000 -0.000150 24000 -0.190000 25000 -0.010000
     26000 -0.701730 28000 -0.092500
M2  1000 -0.077421 6000 -0.922579                    $Polystyrene
c
c Source Definition
SDEF PAR=P axs=0 0 1 pos=0 0 0 rad=d1 ext=d2 erg=d3 cel=10   $Cylindrical source located inside inner cylinder
SI1 0 60.0075
SI2 0.9525 137.4424
SI3 L 0.13 .013 .013 .0532 .0664 .0727 .09 .0933 .105      $Discrete energy distribution based on gamma energies from MicroShield
     .1091 .12 .1214 .1408 .1438 .1633 .1827 .1857 .1904
     .1949 .2021 .2053 .2214
SP3  7.112e7 3.166e7 1.201e7 7.990e5 3.478e5 4.273e4 1.060e6     | $Gamma emission frequency from MicroShield
     1.732e6 8.014e5 5.827e5 5.827e4 2.712e5 8.547e4 4.079e6
     1.826e6 1.554e5 2.098e7 3.572e5 2.292e5 3.885e5 1.826e6
     3.885e4
c
c Tally Definition
F15:p 91.43 0 69.19745 10                           $Point Detector, mid height of vessel, one foot away
F25:p 91.43 243.84 69.19745 10                      $Point Detector, mid height of vessel, one foot away, 96" offset
c                                                   $equivalent to dose contribution from lag vessel
FM15 1.5045072E8                                    $Tally multiplier equivalent to summation of all gamma emission frequencies
FM25 2.2567608e7                                    $Source scaled to one week of buildup in lag vessel
c
c   ANSI 6.1.1-1977 Gamma Flux to Dose Conversion Factors, using US units
c   (rem/hr)/(photons/cm2-s)
df0  IU 1 IC 20                                     $Seperates tally into energy bins, applies conversion factors and sums
```

Attachment B

Email Correspondence

Caleb Trainor

From:	Ja-Kael Luey <jluey@kurion.com>
Sent:	Thursday, December 03, 2015 10:38 AM
To:	Caleb Trainor
Cc:	Charles Beatty; jlux@envpm.com; Ja-Kael Luey
Subject:	RE: <EXTERNAL> RE: Resin Vessel Dimensions

Caleb,

If you are doing dose calculations use the following in lieu of specific information (which may be proprietary from AVANTech but I have not had a chance to check).

- Wall Thickness – 3/8" This is the wall thickness for Kurion's standard 36-inch vessel that has a similar working pressure. It is possible the AVANTech wall thickness is thicker so your dose calculation will be bounding. For dose purposes and alpha, not sure if the liner material will make much of a difference in your calculation.
- Spacing: DWG-M110 is the drawing that I thought had the separation distance on it. Looks like this was not a critical element to have on the drawing; however, based on the scale the distance is edge-to-edge as the gap looks to be the same as the ISM Vessel (which is 48-inches).

Ja-Kael

From: Caleb Trainor [mailto:ctrainor@enercon.com]
Sent: Thursday, December 03, 2015 7:00 AM
To: Ja-Kael Luey <jluey@kurion.com>
Subject: RE: <EXTERNAL> RE: Resin Vessel Dimensions

Thank you for the information, this is much closer to what I need. I am still left with a few questions however. Outer diameter is listed but no inner diameter; a liner is also mentioned but again no thickness or material. I assume the 48" spacing is edge to edge, or 96" on center, correct? It will be easy for me to calculate dose for any other spacing you give me, but a change in spacing is ultimately a crit safety concern, not dose.

From: Ja-Kael Luey [mailto:jluey@kurion.com]
Sent: Wednesday, December 02, 2015 11:22 PM
To: Caleb Trainor <ctrainor@enercon.com>
Cc: Charles Beatty <cbeatty@enercon.com>; jlux@envpm.com; Ja-Kael Luey <jluey@kurion.com>
Subject: <EXTERNAL> RE: Resin Vessel Dimensions

Caleb,

Please find attached the drawing that is the basis for the ISM Vessels used in the design. The spacing between the vessels is 48" from the General Assembly drawing (I can provide the specific reference when I am back in the office Thursday if you do not have the full set). This separation is based on meetings held with Enercon and not a specific-cited document. From a spacing standpoint it would be better if this could be smaller, especially for the BA Unit since it is fit into an enclosure.

Ja-Kael

From: Caleb Trainor [mailto:ctrainor@enercon.com]
Sent: Wednesday, December 02, 2015 6:34 AM

To: Ja-Kael Luey <jluey@kurion.com>
Cc: Charles Beatty <cbeatty@enercon.com>; jlux@envpm.com
Subject: Resin Vessel Dimensions

In support of dose calculations off the resin vessels, I am in need of the vessel dimensions, vessel material, and spacing between the vessels. The design drawings I have so far only detail the piping, and not the vessels. Can you help me get a hold of this information?

Caleb Trainor
Emergency Preparedness Engineer
(o) (813) 962-1800 ext. 207

ENERCON
Excellence—Every project. Every day

Attachment C

MCNP6 Output
Code Name & Version = MCNP6, 1.0

```
  _/      _/        _/_/_/       _/      _/       _/_/_/         _/_/_/
 _/_/  _/_/      _/             _/_/    _/       _/    _/     _/
_/  _/  _/      _/             _/  _/  _/       _/_/_/       _/_/_/
_/      _/      _/             _/    _/_/       _/           _/    _/
_/      _/        _/_/_/       _/      _/       _/             _/_/
```

```
1mcnp     version 6     ld=05/08/13                    12/15/15 16:46:08
****************************************************************************     probid =  12/15/15 16:46:08
i=cim.txt o=cim.out tasks 8

 warning.  Physics models disabled.
        Cimmarron Resin Bed Dose Estimation
        c Cells
        10 2 -0.485139 -100        imp:p=1                    $Resin (poly
        50 1 -8 -10 100          imp:p=2                    $SS 304 laye
        11 1 -8 -101 10 100       imp:p=4                    $SS 304 vess
        30 0 -300 101            imp:p=8                    $Void
        40 0 300               imp:p=0                    $Boundary

        c Surfaces
         100 RCC 0 0 0.9525 0 0 136.4899 60.0075                  $Interior cy
         10  RCC 0 0 0.47625 0 0 136.96615 60.48375                 $Layer used
         101 RCC 0 0 0 0 0 138.3949 60.96                    $Outer cylin
         102 S 91.43 0 69.19745 20                        $Detector 1
         103 S 91.43 243.84 69.19745 20                      $Detector 2
         300 RPP -65 100 -65 250 0 150                      $Bounding Ar

         c Data
         Mode P
         nps 10e6
         c
         c Material Definitions
         M1  6000 -0.000400 14000 -0.005000 15000 -0.000230             $SS 304
             16000 -0.000150 24000 -0.190000 25000 -0.010000
             26000 -0.701730 28000 -0.092500
         M2  1000 -0.077421 6000 -0.922579                    $Polystyrene
         c
         c Source Definition
```

```
SDEF PAR=P axs=0 0 1 pos=0 0 0 rad=d1 ext=d2 erg=d3 cel=10        $Cylindrical
SI1 0 60.0075
SI2 0.9525 137.4424
SI3 L 0.13 .013 .013 .0532 .0664 .0727 .09 .0933 .105            $Discrete en
    .1091 .12 .1214 .1408 .1438 .1633 .1827 .1857 .1904
    .1949 .2021 .2053 .2214
SP3  7.112e7 3.166e7 1.201e7 7.990e5 3.478e5 4.273e4 1.060e6      $Gamma emiss
    1.732e6 8.014e5 5.827e5 5.827e4 2.712e5 8.547e4 4.079e6
    1.826e6 1.554e5 2.098e7 3.572e5 2.292e5 3.885e5 1.826e6
    3.885e4
c
c Tally Definition
F15:p 91.43 0 69.19745 10                         $Point Detec
F25:p 91.43 243.84 69.19745 10                       $Point Detec
c                                           $equivalent
FM15 1.5045072E8                                  $Tally multi
FM25 2.2567608e7                                  $Source scal
c
c  ANSI 6.1.1-1977 Gamma Flux to Dose Conversion Factors, using US units
c  (rem/hr)/(photons/cm2-s)
df0  IU 1 IC 20                                 $Seperates t

surface    100.2 and surface     10.2 are the same.     10.2 will be deleted.

surface    101.3 and surface    300.6 are the same.    300.6 will be deleted.

 comment.        2 surfaces were deleted for being the same as others.

 warning.  1 materials had unnormalized fractions. print table 40.
1cells                                                    print table 60

                 atom       gram                              photon
       cell    mat   density     density     volume      mass         pieces importance
```

```
     1    10      2  4.48803E-02 4.85139E-01 1.54405E+06 7.49079E+05        1  1.0000E+00
     2    50      1  8.76841E-02 8.00000E+00 3.00795E+04 2.40636E+05        1  2.0000E+00
     3    11      1  8.76841E-02 8.00000E+00 4.15670E+04 3.32536E+05        1  4.0000E+00
     4    30      0  0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00        0  8.0000E+00
     5    40      0  0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00        0  0.0000E+00

total                            1.61570E+06 1.32225E+06

 warning.  surface     102.0 is not used for anything.

 warning.  surface     103.0 is not used for anything.

   minimum source weight = 1.0000E+00    maximum source weight = 1.0000E+00

***********************************************************
* Random Number Generator  =                   1 *
* Random Number Seed       =      19073486328125 *
* Random Number Multiplier =      19073486328125 *
* Random Number Adder      =                   0 *
* Random Number Bits Used  =                  48 *
* Random Number Stride     =              152917 *
***********************************************************


 comment.  threading will be used when possible in portions of mcnp6.

 comment.  threading will be used for n/p/e table physics.

 comment.  threading will generally not be used for model physics.

       4 warning messages so far.
1cross-section tables                                              print table 100
   XSDIR used: C:\MCNP\MCNP_DATA/xsdir_mcnp6.1
```

table length

tables from file xdata/mcplib84

1000.84p	1974	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
6000.84p	3228	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
14000.84p	4868	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
15000.84p	4574	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
16000.84p	4730	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
24000.84p	5758	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
25000.84p	5674	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
26000.84p	5794	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
28000.84p	5902	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12

total 42502

maximum photon energy set to 100.0 mev (maximum electron energy)

tables from file xdata/el03

1000.03e	2329	6/6/98
6000.03e	2333	6/6/98
14000.03e	2339	6/6/98
15000.03e	2339	6/6/98
16000.03e	2339	6/6/98
24000.03e	2345	6/6/98
25000.03e	2345	6/6/98
26000.03e	2345	6/6/98
28000.03e	2347	6/6/98

1particles and energy limits print table 101

particle cutoff	maximum particle	smallest table	largest table	always use table	always use model

```
particle type      energy      energy      maximum      maximum      below      above

 2 p   photon    1.0000E-03  1.0000E+02  1.0000E+05  1.0000E+05  1.0000E+36  1.0000E+36
 3 e   electron  1.0000E-03  1.0000E+02  1.0000E+02  1.0000E+02  1.0000E+36  1.0000E+36


warning.  material      1 has been set to a conductor.

*******************************************************************************************************************

dump no.    1 on file runtpe    nps =         0    coll =            0    ctm =       0.00   nrn =
  0

     5 warning messages so far.

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
 1 4.2865E-04 2.5000E-01 1.4603E+01 3.0014E-01 3.1687E+01 1.0000E+01 1.0341E-01   11    29113   6              117       6

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
 1 4.6727E-04 2.5000E-01 2.9341E+01 6.9818E-01 3.5255E+01 1.0000E+01 1.0449E-01   11    37886   2               36       6

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
 1 4.3285E-04 2.5000E-01 2.9375E+01 7.7426E-01 3.5284E+01 1.0000E+01 1.0449E-01   11    37886   4               67       6

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
 1 7.1396E-04 2.5000E-01 2.9074E+01 3.7427E-01 3.3383E+01 1.0000E+01 1.8570E-01   11    99391   3               45       6

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
 1 5.0577E-04 2.5000E-01 7.5862E+01 1.0924E+00 4.4741E+01 1.0000E+01 1.3005E-01   11    202920   4              79       6

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
 1 5.2451E-04 2.5000E-01 1.8136E+01 4.9620E-02 3.6183E+01 1.0000E+01 1.3000E-01   11    266011   8             111       6

det    t       wgt       psc      amfp      ddetx     radius      erg     cell       nps  nch  p         nrn     ipsc
```

```
 2 7.3518E-05 2.5000E-01 8.4093E+01 8.6295E-02 2.0433E+02 1.0000E+01 1.1394E-01   11    267347   4          66     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 1.6404E-03 1.0000E+00 2.5645E+02 2.3748E+00 4.8113E+01 1.0000E+01 1.7341E-01   10    275363   3          54     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 1.4661E-03 2.5000E-01 6.2765E+01 5.1569E-01 3.1892E+01 1.0000E+01 1.1387E-01   11    289480   3          65     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 4.7999E-04 2.5000E-01 4.1311E+01 1.1433E+00 3.3039E+01 1.0000E+01 9.5904E-02   11    295930   3          60     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 5.3206E-03 2.5000E-01 1.4027E+02 7.2495E-02 3.1234E+01 1.0000E+01 1.8570E-01   11    313718   3          41     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 2 7.5842E-05 2.5000E-01 8.3107E+01 1.0047E-02 2.0776E+02 1.0000E+01 1.0709E-01   11    385577   4          71     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 2 1.4009E-04 2.5000E-01 2.5325E+02 4.0166E-01 2.1940E+02 1.0000E+01 1.7177E-01   11    388463   5          65     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 4.8092E-04 2.5000E-01 1.5055E+01 1.2837E-01 3.3098E+01 1.0000E+01 1.1808E-01   11    397905   3          61     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 2 6.9698E-05 5.0000E-01 1.3953E+02 1.3767E+00 2.0052E+02 1.0000E+01 1.3000E-01   50    571314   2          42     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 5.6973E-04 2.5000E-01 7.3557E+01 1.4935E+00 3.3967E+01 1.0000E+01 9.3916E-02   11    573255   6          88     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 2.9775E-03 2.5000E-01 8.8545E+01 1.9940E-01 3.1134E+01 1.0000E+01 1.0968E-01   11    674671  13         254     6

det   t      wgt       psc      amfp      ddetx    radius      erg     cell       nps  nch   p          nrn     ipsc
 1 3.4525E-03 2.5000E-01 1.1223E+02 1.9960E-01 3.2548E+01 1.0000E+01 1.3000E-01   11    717579   2          48     6
```

```
det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 6.2456E-04 5.0000E-01 3.2244E+01 1.4263E+00 3.1413E+01 1.0000E+01 1.2246E-01   50    726168   3     45    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 2 1.6632E-04 1.0000E+00 2.2685E+02 1.6681E+00 2.0234E+02 1.0000E+01 1.4318E-01   10    812481   3     60    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 5.3368E-04 1.0000E+00 8.8019E+01 2.6691E+00 4.2655E+01 1.0000E+01 1.3000E-01   10    860193   1     13    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 2 8.1664E-05 2.5000E-01 1.1185E+02 2.6078E-01 2.0491E+02 1.0000E+01 1.8570E-01   11    864981   1     21    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 1.5431E-03 2.5000E-01 5.5340E+01 2.4316E-01 3.3450E+01 1.0000E+01 1.0500E-01   11    874440   6    101    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 5.2532E-04 2.5000E-01 1.5099E+01 4.8062E-02 3.3015E+01 1.0000E+01 1.3000E-01   11    911985   1     15    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 4.8100E-04 2.5000E-01 1.5132E+01 1.3574E-01 3.3059E+01 1.0000E+01 1.3000E-01   11    911985   2     23    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 1.5455E-03 2.5000E-01 7.6267E+01 5.5582E-01 3.3559E+01 1.0000E+01 9.8397E-02   11    959316   5     95    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 2 6.8770E-05 2.5000E-01 9.9724E+01 2.9810E-01 2.0694E+02 1.0000E+01 1.2357E-01   11    991642   2     37    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 1.9707E-03 2.5000E-01 7.4492E+01 1.5406E-01 3.5907E+01 1.0000E+01 1.4937E-01   11   1035243   4     59    6

det   t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p         nrn   ipsc
 1 1.1657E-03 2.5000E-01 7.1482E+01 6.2134E-01 3.6204E+01 1.0000E+01 1.4937E-01   11   1035243   5     67    6
```

```
det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 6.7144E-04 5.0000E-01 6.2578E+01 1.6644E+00 3.7469E+01 1.0000E+01 1.4380E-01   50    1078642   5       86    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 2 9.4945E-05 2.5000E-01 1.7751E+02 5.5729E-01 2.0641E+02 1.0000E+01 1.8018E-01   11    1087746   3       65    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 8.7006E-04 2.5000E-01 2.9759E+01 1.4314E-01 3.4342E+01 1.0000E+01 1.2527E-01   11    1111829   7      110    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 4.7657E-04 2.5000E-01 2.9136E+01 7.0819E-01 3.4614E+01 1.0000E+01 1.2527E-01   11    1111829   9      129    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 1.4361E-03 2.5000E-01 8.6140E+01 2.7980E-01 4.2475E+01 1.0000E+01 1.2024E-01   11    1169084   4       84    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 5.0758E-04 2.5000E-01 2.4111E+01 2.5089E-01 3.8349E+01 1.0000E+01 1.1081E-01   11    1204350   5       99    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 7.1728E-04 5.0000E-01 9.5580E+01 1.9686E+00 3.8483E+01 1.0000E+01 1.4930E-01   50    1232639   3       59    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 2 7.5601E-05 5.0000E-01 1.0312E+02 9.9309E-01 2.0052E+02 1.0000E+01 1.3000E-01   50    1318369   1       17    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 5.6998E-04 2.5000E-01 4.5313E+01 9.8667E-01 3.4341E+01 1.0000E+01 1.0458E-01   11    1385745   4       88    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 2 6.6728E-05 1.0000E+00 9.3301E+01 1.6266E+00 2.0916E+02 1.0000E+01 1.6869E-01   10    1395745   3       70    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
 1 5.5334E-04 2.5000E-01 1.5005E+01 8.8175E-02 3.1430E+01 1.0000E+01 1.3375E-01   11    1430896   3       46    6

det   t         wgt         psc         amfp        ddetx       radius      erg         cell        nps   nch  p          nrn    ipsc
```

```
 1 1.0282E-03 5.0000E-01 5.9984E+01 1.4429E+00 3.3116E+01 1.0000E+01 1.3000E-01   50    1572051   6       108    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 2 6.6336E-05 2.5000E-01 8.0864E+01 1.8158E-01 2.0112E+02 1.0000E+01 1.0217E-01   11    1588704   4        85    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 5.0668E-04 2.5000E-01 2.5289E+01 5.6608E-01 3.3578E+01 1.0000E+01 1.7126E-01   11    1655950   7       102    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 6.3800E-04 2.5000E-01 2.5121E+01 3.3871E-01 3.3414E+01 1.0000E+01 1.7126E-01   11    1655950   8       110    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 8.5434E-04 2.5000E-01 2.4413E+01 1.4079E-01 3.1427E+01 1.0000E+01 1.8570E-01   11    1664001   3        39    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 4.9346E-04 2.5000E-01 1.9930E+01 1.3594E-01 3.7453E+01 1.0000E+01 1.8570E-01   11    1695063   1        20    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 1.1612E-03 5.0000E-01 1.2665E+02 1.8239E+00 3.7428E+01 1.0000E+01 1.4380E-01   50    1771399   6        99    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 5.2350E-04 2.5000E-01 7.0521E+01 1.5161E+00 3.4306E+01 1.0000E+01 9.8239E-02   11    1782361   16      292    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 2 9.3573E-05 2.5000E-01 1.0677E+02 5.8288E-02 2.0695E+02 1.0000E+01 1.1075E-01   11    1800281   2        40    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 5.0019E-04 2.5000E-01 3.6268E+01 4.2032E-01 4.3531E+01 1.0000E+01 1.2915E-01   11    1811526   13      236    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 1 5.2333E-04 2.5000E-01 2.9283E+01 2.1129E-01 4.2454E+01 1.0000E+01 1.7037E-01   11    1833593   8       136    6

det   t      wgt     psc     amfp     ddetx    radius     erg    cell       nps  nch  p          nrn    ipsc
 2 1.5870E-04 5.0000E-01 1.6024E+02 6.9714E-01 2.0004E+02 1.0000E+01 1.7501E-01   50    1869062   2        38    6
```

```
det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 7.5992E-04 2.5000E-01 4.3218E+01 7.8143E-01 3.2184E+01 1.0000E+01 1.0812E-01  11   1925119  4      58   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 9.3520E-04 2.5000E-01 3.2528E+01 9.9967E-02 3.5387E+01 1.0000E+01 1.3000E-01  11   1925442  2      26   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 2 1.5937E-04 2.5000E-01 2.2703E+02 3.1759E-01 2.0312E+02 1.0000E+01 1.8570E-01  11   1956285  8     105   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 4.7675E-04 5.0000E-01 2.4336E+01 1.2345E+00 3.4381E+01 1.0000E+01 1.8570E-01  50   2159382  3      40   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 2.0617E-03 5.0000E-01 9.7782E+01 1.0548E+00 3.6255E+01 1.0000E+01 1.4656E-01  50   2166494  2      40   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 5.4699E-04 5.0000E-01 9.1756E+01 2.0320E+00 4.1829E+01 1.0000E+01 1.4091E-01  50   2183252  5      80   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 6.1567E-04 2.5000E-01 5.8573E+01 1.0186E+00 3.6971E+01 1.0000E+01 9.2406E-02  11   2319969  7     124   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 1.2852E-03 2.5000E-01 1.3956E+02 7.2292E-01 4.5792E+01 1.0000E+01 1.8570E-01  11   2334214  2      44   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 1.1976E-03 5.0000E-01 5.3714E+01 1.2483E+00 3.2005E+01 1.0000E+01 1.2771E-01  50   2355712  4      71   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 1.1893E-03 5.0000E-01 8.8997E+01 1.4704E+00 3.6995E+01 1.0000E+01 1.2044E-01  50   2443068  2      41   6

det    t       wgt      psc      amfp     ddetx    radius    erg     cell      nps  nch  p       nrn    ipsc
 1 1.6740E-03 5.0000E-01 1.2254E+02 1.6349E+00 3.3701E+01 1.0000E+01 1.3000E-01  50   2452650  5      92   6
```

```
det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 2 6.1736E-05 2.5000E-01 1.1410E+02 5.3949E-01 2.0706E+02 1.0000E+01 1.4751E-01  11  2497606  13   209  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 8.9798E-04 2.5000E-01 6.1676E+01 8.2533E-01 3.4601E+01 1.0000E+01 1.3000E-01  11  2556120  2   46  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 5.2836E-04 2.5000E-01 1.9075E+01 1.7225E-01 3.4773E+01 1.0000E+01 1.1794E-01  11  2577928  3   55  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 3.2627E-03 5.0000E-01 1.9452E+02 9.4308E-01 4.2983E+01 1.0000E+01 1.7960E-01  50  2625505  4   60  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 5.3011E-04 2.5000E-01 6.7960E+01 7.7162E-01 4.8559E+01 1.0000E+01 1.4380E-01  11  2647549  7   90  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 2 8.0678E-05 5.0000E-01 1.8015E+02 1.4568E+00 2.0347E+02 1.0000E+01 1.4380E-01  50  2729735  2   25  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 5.1116E-04 5.0000E-01 4.2447E+01 1.5137E+00 3.8136E+01 1.0000E+01 1.3000E-01  50  2780623  5   82  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 6.2670E-04 2.5000E-01 2.0365E+01 2.5639E-01 3.1631E+01 1.0000E+01 1.3000E-01  11  2878562  2   24  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 6.8741E-04 2.5000E-01 9.0621E+01 9.0706E-01 4.6017E+01 1.0000E+01 1.4380E-01  11  2942792  11   207  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 3.2928E-03 2.5000E-01 1.1130E+02 2.9941E-01 3.1574E+01 1.0000E+01 1.7372E-01  11  3091738  2   39  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
 1 6.2637E-04 2.5000E-01 2.9437E+01 5.2772E-01 3.3214E+01 1.0000E+01 1.3000E-01  11  3116909  6   102  6

det  t  wgt  psc  amfp  ddetx  radius  erg  cell  nps nch p  nrn  ipsc
```

```
 1 5.8600E-04 2.5000E-01 7.7180E+01 1.0494E+00 4.2836E+01 1.0000E+01 1.3000E-01  11   3134216  7          87    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 4.8181E-04 2.5000E-01 3.0576E+01 2.9797E-01 4.3294E+01 1.0000E+01 1.3000E-01  11   3161023  1          15    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 5.4533E-04 2.5000E-01 4.4832E+01 1.0679E+00 3.3531E+01 1.0000E+01 1.0656E-01  11   3185384  9          161   6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 2 6.3824E-05 2.5000E-01 7.9431E+01 1.8690E-01 2.0267E+02 1.0000E+01 1.3000E-01  11   3185455  3          35    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 5.9247E-04 2.5000E-01 3.7973E+01 6.9944E-01 3.5596E+01 1.0000E+01 1.2844E-01  11   3195092  2          36    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 8.7236E-04 2.5000E-01 5.6743E+01 7.8042E-01 3.4437E+01 1.0000E+01 1.3000E-01  11   3220944  11         193   6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 5.7987E-04 2.5000E-01 6.0033E+01 1.0150E+00 3.8636E+01 1.0000E+01 1.2286E-01  11   3245009  2          36    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 4.6707E-04 5.0000E-01 3.4993E+01 1.3552E+00 3.9211E+01 1.0000E+01 1.6379E-01  50   3276500  2          36    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 4.8260E-04 5.0000E-01 3.1520E+01 1.4794E+00 3.4406E+01 1.0000E+01 1.2549E-01  50   3347693  4          71    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 5.1747E-04 5.0000E-01 4.9909E+01 1.4486E+00 4.2460E+01 1.0000E+01 1.3669E-01  50   3357499  4          90    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 1.2658E-03 5.0000E-01 8.5492E+01 1.4898E+00 3.4807E+01 1.0000E+01 1.3000E-01  50   3361939  1          16    6

det   t       wgt      psc      amfp      ddetx     radius     erg    cell      nps  nch  p      nrn    ipsc
 1 1.3289E-03 2.5000E-01 8.5079E+01 7.6544E-01 3.4421E+01 1.0000E+01 1.3000E-01  11   3361939  5          57    6
```

```
det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 1.1435E-03 2.5000E-01 7.2440E+01 4.1929E-01 4.0711E+01 1.0000E+01 1.2771E-01  11   3367095  2           34     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 8.5953E-04 5.0000E-01 8.8525E+01 2.0934E+00 3.1785E+01 1.0000E+01 1.0117E-01  50   3394891  2           41     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 2 7.4288E-05 5.0000E-01 9.8099E+01 9.3087E-01 2.0353E+02 1.0000E+01 1.8570E-01  50   3442182  2           39     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 1.3308E-03 2.5000E-01 1.2372E+02 8.8648E-01 3.9043E+01 1.0000E+01 1.5373E-01  11   3537462  12          232     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 5.0014E-04 2.5000E-01 3.3212E+01 2.3959E-01 4.5599E+01 1.0000E+01 1.3000E-01  11   3539926  2           27     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 5.3956E-04 5.0000E-01 2.1833E+01 1.1026E+00 3.2697E+01 1.0000E+01 1.8570E-01  50   3545089  1           14     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 5.6292E-04 2.5000E-01 1.6586E+01 1.1803E-01 3.2278E+01 1.0000E+01 1.8283E-01  11   3650364  2           35     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 2.2950E-03 2.5000E-01 9.2796E+01 3.3100E-01 3.3992E+01 1.0000E+01 1.3000E-01  11   3650975  3           33     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 2 9.7066E-05 1.0000E+00 2.6143E+02 2.2765E+00 2.0976E+02 1.0000E+01 1.3000E-01  10   3654321  1           16     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 1.1695E-03 2.5000E-01 5.1011E+01 6.0644E-01 3.0763E+01 1.0000E+01 1.1012E-01  11   3848369  4           73     6

det   t        wgt        psc        amfp       ddetx      radius     erg        cell       nps  nch  p       nrn    ipsc
 1 6.6962E-04 2.5000E-01 7.6065E+01 8.3818E-01 4.4213E+01 1.0000E+01 1.7413E-01  11   3905496  4           71     6
```

```
det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 2 9.6599E-05 2.5000E-01 1.1998E+02 1.0080E-01 2.1137E+02 1.0000E+01 1.8479E-01  11   3906556  2      32   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 2 6.7549E-05 2.5000E-01 9.7349E+01 2.9342E-01 2.0679E+02 1.0000E+01 1.3000E-01  11   3907053  6     109   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 2 7.7800E-05 2.5000E-01 1.0680E+02 3.0907E-01 2.0024E+02 1.0000E+01 1.0923E-01  11   3921665  7     112   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 8.6059E-04 1.0000E+00 4.1885E+01 1.6285E+00 3.8987E+01 1.0000E+01 1.8570E-01  10   3930467  1      13   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 3.2872E-03 2.5000E-01 2.5142E+02 7.6437E-01 3.7643E+01 1.0000E+01 1.8570E-01  11   4059754  2      41   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 5.4959E-04 5.0000E-01 7.4183E+01 2.1380E+00 3.5585E+01 1.0000E+01 1.2589E-01  50   4079838  3      59   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 9.3644E-04 5.0000E-01 4.5728E+01 1.0991E+00 3.5982E+01 1.0000E+01 1.8570E-01  50   4156195  1      14   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 7.2649E-04 5.0000E-01 4.6224E+01 1.3520E+00 3.6194E+01 1.0000E+01 1.8570E-01  50   4156195  5      69   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 1.3817E-03 2.5000E-01 5.9445E+01 3.0319E-01 3.5554E+01 1.0000E+01 9.3136E-02  11   4159952  6     103   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 2.0539E-03 5.0000E-01 1.1802E+02 1.2470E+00 3.6250E+01 1.0000E+01 1.3000E-01  50   4171884  1      16   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
 1 5.3407E-04 5.0000E-01 4.4289E+01 1.3847E+00 4.0649E+01 1.0000E+01 1.3000E-01  50   4182657  1      14   6

det   t         wgt       psc       amfp      ddetx     radius    erg       cell      nps  nch  p    nrn   ipsc
```

```
 1 6.2597E-04 5.0000E-01 5.3306E+01 1.6796E+00 3.5545E+01 1.0000E+01 1.5492E-01  50   4265308  2      35    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 2 1.2845E-04 2.5000E-01 1.3873E+02 4.2095E-02 2.0298E+02 1.0000E+01 1.8570E-01  11   4310093  4      64    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 7.0270E-04 2.5000E-01 2.8470E+01 3.4209E-01 3.3838E+01 1.0000E+01 1.3000E-01  11   4328280  2      23    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 6.2254E-04 5.0000E-01 7.6286E+01 1.5238E+00 4.6095E+01 1.0000E+01 1.4380E-01  50   4354629  1      14    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 5.1419E-04 5.0000E-01 1.9935E+01 9.4154E-01 3.4688E+01 1.0000E+01 2.0530E-01  50   4398274  1      14    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 1.3410E-03 1.0000E+00 2.7016E+02 2.7793E+00 4.4615E+01 1.0000E+01 1.2317E-01  10   4535430  2      36    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 1.9512E-03 5.0000E-01 1.2248E+02 1.6198E+00 3.1444E+01 1.0000E+01 1.2842E-01  50   4573242  9     197    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 2 1.0099E-04 5.0000E-01 1.9850E+02 1.2931E+00 2.0718E+02 1.0000E+01 1.8570E-01  50   4594708  2      24    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 6.2792E-04 5.0000E-01 4.3828E+01 1.4149E+00 3.6735E+01 1.0000E+01 1.1670E-01  50   4608539  2      41    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 4.7662E-04 1.0000E+00 1.4555E+02 3.4950E+00 3.8406E+01 1.0000E+01 1.0012E-01  10   4613891  3      62    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 1 6.7040E-04 5.0000E-01 4.4527E+01 1.4417E+00 3.5358E+01 1.0000E+01 1.3000E-01  50   4681007  5      52    6

det   t     wgt     psc     amfp     ddetx    radius     erg    cell      nps  nch  p      nrn    ipsc
 2 7.3379E-05 1.0000E+00 7.0702E+01 1.2945E+00 2.0499E+02 1.0000E+01 1.8570E-01  10   4684360  1      16    6
```

```
det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 1 1.0558E-03 2.5000E-01 4.6122E+01 2.6416E-01 3.6532E+01 1.0000E+01 1.3000E-01  11    4703340   2          32      6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 1 1.5123E-03 2.5000E-01 8.6092E+01 8.4980E-01 3.1118E+01 1.0000E+01 1.3000E-01  11    4712733   2          23      6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 1 1.7255E-03 2.5000E-01 8.5665E+01 7.1743E-01 3.1048E+01 1.0000E+01 1.3000E-01  11    4712733   7          97      6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 1 2.9430E-03 2.5000E-01 1.1299E+02 2.5841E-01 3.4348E+01 1.0000E+01 1.4855E-01  11    4764056   8          137     6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 1 6.1087E-04 2.5000E-01 2.3456E+01 3.8734E-02 3.8338E+01 1.0000E+01 1.1820E-01  11    4768139   6          100     6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 2 8.9505E-05 2.5000E-01 9.3860E+01 3.4463E-02 2.0078E+02 1.0000E+01 1.8570E-01  11    4879138   1          15      6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 2 9.3422E-05 2.5000E-01 1.7357E+02 3.4902E-01 2.2835E+02 1.0000E+01 1.6992E-01  11    4985150   7          106     6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 2 7.9443E-05 2.5000E-01 1.3739E+02 5.1180E-01 2.0309E+02 1.0000E+01 1.2469E-01  11    5166101   16         311     6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 2 6.3427E-05 5.0000E-01 1.4501E+02 1.3501E+00 2.1716E+02 1.0000E+01 1.8570E-01  50    5240953   1          23      6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 2 9.2058E-05 2.5000E-01 1.2053E+02 2.2794E-01 2.0366E+02 1.0000E+01 1.8570E-01  11    5309294   1          17      6

det   t        wgt        psc        amfp       ddetx      radius      erg      cell        nps  nch   p        nrn    ipsc
 2 8.1922E-05 2.5000E-01 1.0587E+02 2.1140E-01 2.0402E+02 1.0000E+01 1.3000E-01  11    5507091   5          73      6
```

```
det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 7.1386E-05 2.5000E-01 1.6026E+02 6.6252E-01 2.1460E+02 1.0000E+01 1.6330E-01  11   5607881  4          52    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 1.2320E-04 5.0000E-01 1.5423E+02 8.2608E-01 2.0883E+02 1.0000E+01 1.6773E-01  50   5742616  4          95    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 6.7748E-05 2.5000E-01 1.4851E+02 6.1173E-01 2.1751E+02 1.0000E+01 1.6067E-01  11   5798661  6          98    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 1.4616E-04 5.0000E-01 2.5270E+02 1.1450E+00 2.0924E+02 1.0000E+01 1.8570E-01  50   5941969  1          15    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 6.3250E-05 5.0000E-01 1.0746E+02 1.1696E+00 2.0489E+02 1.0000E+01 1.5957E-01  50   6106548  3          63    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 7.7663E-05 2.5000E-01 1.1530E+02 3.6248E-01 2.0275E+02 1.0000E+01 1.3000E-01  11   6122751  3          40    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 8.2003E-05 5.0000E-01 8.5367E+01 6.7714E-01 2.0516E+02 1.0000E+01 1.7525E-01  50   6128781  6         107    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 8.0096E-05 2.5000E-01 8.1642E+01 1.4839E-02 1.9990E+02 1.0000E+01 1.3000E-01  11   6301952  1          16    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 7.7647E-05 2.5000E-01 1.1386E+02 2.1132E-01 2.1732E+02 1.0000E+01 1.6915E-01  11   6325708  2          44    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 8.0105E-05 2.5000E-01 9.7521E+01 9.2966E-02 2.1009E+02 1.0000E+01 1.2952E-01  11   6569431  8         143    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
 2 6.3612E-05 5.0000E-01 9.4114E+01 1.0530E+00 2.0267E+02 1.0000E+01 1.3000E-01  50   6864922  1          15    6

det  t      wgt       psc       amfp      ddetx     radius    erg       cell      nps nch p       nrn  ipsc
```

```
 2 8.5899E-05 2.5000E-01 9.9501E+01 5.7195E-02 2.0863E+02 1.0000E+01 1.3000E-01  11   6957752  3    47   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 8.3686E-05 2.5000E-01 1.8246E+02 6.7611E-01 2.1005E+02 1.0000E+01 1.8570E-01  11   6982788  1    22   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 1.4716E-04 5.0000E-01 1.8785E+02 8.4998E-01 2.0837E+02 1.0000E+01 1.4865E-01  50   7006398  5    97   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 8.0744E-05 1.0000E+00 2.2806E+02 2.3206E+00 2.1012E+02 1.0000E+01 1.3000E-01  10   7015682  1    14   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 7.7876E-05 2.5000E-01 9.8330E+01 2.0527E-01 2.0228E+02 1.0000E+01 1.0346E-01  11   7071106  7    116  6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 7.8800E-05 2.5000E-01 1.1116E+02 2.6310E-01 2.0772E+02 1.0000E+01 1.4380E-01  11   7237656  3    44   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 8.2322E-05 2.5000E-01 9.5347E+01 6.8997E-02 2.0739E+02 1.0000E+01 1.0388E-01  11   7430193  8    150  6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 7.0185E-05 5.0000E-01 1.4111E+02 1.3704E+00 2.0159E+02 1.0000E+01 1.2327E-01  50   7480406  3    46   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 1.9834E-04 5.0000E-01 2.8574E+02 9.6830E-01 2.0865E+02 1.0000E+01 2.0210E-01  50   7980777  6    106  6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 8.0021E-05 2.5000E-01 1.0923E+02 2.7107E-01 2.0351E+02 1.0000E+01 1.3000E-01  11   7983577  6    88   6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 1.0526E-04 2.5000E-01 1.3134E+02 4.5678E-02 2.1778E+02 1.0000E+01 1.2624E-01  11   8224664  6    117  6

det  t      wgt     psc     amfp     ddetx    radius     erg    cell     nps  nch  p     nrn   ipsc
 2 8.7746E-05 2.5000E-01 1.0136E+02 7.5749E-02 2.0642E+02 1.0000E+01 1.8570E-01  11   8660330  2    35   6
```

```
det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 8.7664E-05 2.5000E-01 1.1636E+02 7.3317E-02 2.2154E+02 1.0000E+01 1.6760E-01  11    8932238   8      138    6

det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 7.1299E-05 5.0000E-01 1.2433E+02 1.2120E+00 2.0321E+02 1.0000E+01 1.2898E-01  50    9142262   5       67    6

det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 6.6424E-05 2.5000E-01 9.6398E+01 3.4061E-01 2.0267E+02 1.0000E+01 1.0685E-01  11    9325623  14      248    6

det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 7.6836E-05 2.5000E-01 9.5882E+01 2.1270E-01 2.0034E+02 1.0000E+01 1.0768E-01  11    9602391  10      167    6

det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 1.0410E-04 2.5000E-01 1.4581E+02 2.4234E-01 2.0913E+02 1.0000E+01 1.4970E-01  11    9649833   4       82    6

det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 7.4218E-05 2.5000E-01 8.0648E+01 7.2804E-02 2.0050E+02 1.0000E+01 1.4349E-01  11    9681295   5       94    6

det  t          wgt        psc        amfp       ddetx      radius     erg        cell       nps nch  p     nrn   ipsc
 2 6.6683E-05 5.0000E-01 1.1436E+02 1.1936E+00 2.0340E+02 1.0000E+01 1.3000E-01  50    9952165   1       16    6
1problem summary

   run terminated when    10000000  particle histories were done.
+                                                 12/15/15 16:50:13
   Cimmarron Resin Bed Dose Estimation                     probid =  12/15/15 16:46:08

 photon creation     tracks      weight        energy         photon loss        tracks      weight        energy
                              (per source particle)                                     (per source particle)

 source            10000000    1.0000E+00    1.0453E-01        escape            1609553    4.0239E-02    5.5397E-03
 nucl. interaction        0    0.            0.                energy cutoff           0    0.            4.5751E-06
 particle decay           0    0.            0.                time cutoff             0    0.            0.
 weight window            0    0.            0.                weight window           0    0.            0.
```

```
cell importance   8402371  2.0823E-02  1.8945E-03      cell importance    557774  2.0827E-02  1.8943E-03
weight cutoff           0  0.          0.              weight cutoff           0  0.          0.
e or t importance       0  0.          0.              e or t importance       0  0.          0.
dxtran                  0  0.          0.              dxtran                  0  0.          0.
forced collisions       0  0.          0.              forced collisions       0  0.          0.
exp. transform          0  0.          0.              exp. transform          0  0.          0.
from neutrons           0  0.          0.              compton scatter         0  0.          5.0425E-02
bremsstrahlung     289865  1.2283E-02  1.1847E-04      capture          20289522  1.1232E+00  4.9640E-02
p-annihilation          0  0.          0.              pair production         0  0.          0.
photonuclear            0  0.          0.              photonuclear abs        0  0.          0.
electron x-rays         0  0.          0.              loss to photofis        0  0.          0.
compton fluores         0  0.          0.
muon capt fluores       0  0.          0.
1st fluorescence  3764613  1.5121E-01  9.6138E-04
2nd fluorescence        0  0.          0.
(gamma,xgamma)          0  0.          0.
tabular sampling        0  0.          0.
prompt photofis         0  0.          0.
  total          22456849  1.1843E+00  1.0750E-01        total          22456849  1.1843E+00  1.0750E-01

  number of photons banked               7207156    average time of (shakes)         cutoffs
  photon tracks per source particle   2.2457E+00      escape          2.1082E-01     tco   1.0000E+33
  photon collisions per source particle 7.3644E+00    capture         1.5924E-01     eco   1.0000E-03
  total photon collisions               73644179    capture or escape 1.6103E-01     wc1  -5.0000E-01
                                                    any termination   1.6002E-01     wc2  -2.5000E-01

computer time so far in this run  236.73 minutes       maximum number ever in bank       5
computer time in mcrun            236.60 minutes       bank overflows to backup file     0
source particles per minute       4.2266E+04
random numbers generated          1463531465        most random numbers used was     1114 in history     3631177

range of sampled source weights = 1.0000E+00 to 1.0000E+00

source efficiency = 1.0000  in cell       10
```

number of histories processed by each thread
1312074 1201267 1259672 1299659 1294866 1282904 1264634 1084924
1photon activity in each cell print table 126

	cell	tracks entering	population	collisions	collisions * weight (per history)	number weighted energy	flux weighted energy	average track weight (relative)	average track mfp (cm)
1	10	10322059	10193584	50738870	5.0739E+00	8.3033E-02	8.3033E-02	1.0000E+00	1.1545E+01
2	50	9023654	8922895	13144162	6.5721E-01	1.1558E-01	1.1558E-01	1.0000E+00	4.3220E-01
3	11	6437568	6528077	9761147	2.4403E-01	1.2524E-01	1.2524E-01	1.0000E+00	4.8793E-01
4	30	1376517	1376517	0	0.0000E+00	1.3735E-01	1.3735E-01	2.0000E+00	0.0000E+00
	total	27159798	27021073	73644179	5.9751E+00				

1tally 15 nps = 10000000
tally type 5 particle flux at a point detector.
particle(s): photons
this tally is modified by standard dose function 1.

this tally is all multiplied by 1.50451E+08

detector located at x,y,z = 9.14300E+01 0.00000E+00 6.91975E+01
2.39237E-05 0.0055

detector located at x,y,z = 9.14300E+01 0.00000E+00 6.91975E+01
uncollided photon flux
7.28378E-06 0.0018

detector score diagnostics

times average score	cumulative fraction of transmissions	transmissions	cumulative tally per history	fraction of total tally
1.00000E-01	2066474	0.33337	9.19114E-09	0.02066

1.00000E+00	2782476	0.78225	4.22849E-08	0.11571
2.00000E+00	480224	0.85973	3.02936E-08	0.18381
5.00000E+00	443038	0.93120	6.23896E-08	0.32405
1.00000E+01	215883	0.96603	6.75746E-08	0.47595
1.00000E+02	202626	0.99871	1.84876E-07	0.89153
1.00000E+03	2350	0.99909	2.79066E-08	0.95426
1.00000E+38	195	0.99913	2.03054E-08	0.99991
before dd roulette	5422	1.00000	4.07592E-11	1.00000

average tally per history = 4.44862E-07 largest score = 5.32061E-03
(largest score)/(average tally) = 1.19601E+04 nps of largest score = 313718

score contributions by cell

	cell	misses	hits	tally per history	weight per hit
1	10	50103740	5496658	2.24457E-07	4.08352E-07
2	50	6049166	355174	8.06515E-08	2.27076E-06
3	11	5057541	346856	1.39754E-07	4.02916E-06
	total	61210447	6198688	4.44862E-07	7.17672E-07

score misses

russian roulette on pd	0
psc=0.	975368
russian roulette in transmission	56695829
underflow in transmission	3539250
hit a zero-importance cell	0
energy cutoff	0

==
====================

results of 10 statistical checks for the estimated answer for the tally fluctuation chart (tfc) bin of tally 15

```
tfc bin   --mean--   ---------relative error---------   ----variance of the variance----   --figure of merit--   -pdf-
behavior  behavior   value  decrease  decrease rate      value  decrease  decrease rate       value    behavior   slope

desired    random    <0.05    yes    1/sqrt(nps)    <0.10    yes     1/nps       constant   random    >3.00
observed   random     0.01    yes       yes          0.02    yes      yes        constant   random     3.12
passed?     yes       yes     yes       yes          yes     yes      yes          yes        yes       yes

===========================================================================================================
=====================
```

this tally meets the statistical criteria used to form confidence intervals: check the tally fluctuation chart to verify.
the results in other bins associated with this tally may not meet these statistical criteria.

----- estimated confidence intervals: -----

estimated asymmetric confidence interval(1,2,3 sigma): 2.3800E-05 to 2.4061E-05; 2.3669E-05 to 2.4192E-05; 2.3538E-05 to 2.4323E-05
estimated symmetric confidence interval(1,2,3 sigma): 2.3793E-05 to 2.4055E-05; 2.3662E-05 to 2.4185E-05; 2.3531E-05 to 2.4316E-05

1analysis of the results in the tally fluctuation chart bin (tfc) for tally 15 with nps = 10000000 print table 160

```
normed average tally per history  = 2.39237E-05        unnormed average tally per history  = 2.39237E-05
estimated tally relative error    = 0.0055           estimated variance of the variance  = 0.0190
relative error from zero tallies  = 0.0005           relative error from nonzero scores  = 0.0055

number of nonzero history tallies =    3040522        efficiency for the nonzero tallies  = 0.3041
history number of largest  tally  =     313718        largest  unnormalized history tally = 3.74298E-01
(largest  tally)/(average tally)  = 1.56455E+04       (largest  tally)/(avg nonzero tally)= 4.75705E+03

(confidence interval shift)/mean  = 0.0003            shifted confidence interval center  = 2.39305E-05
```

if the largest history score sampled so far were to occur on the next history, the tfc bin quantities would change as follows:

estimated quantities	value at nps	value at nps+1	value(nps+1)/value(nps)-1.
mean	2.39237E-05	2.39611E-05	0.001564
relative error	5.47129E-03	5.68168E-03	0.038452
variance of the variance	1.90457E-02	2.19879E-02	0.154486
shifted center	2.39305E-05	2.39314E-05	0.000038
figure of merit	1.41191E+02	1.30928E+02	-0.072686

the estimated inverse power slope of the 200 largest tallies starting at 2.58402E-02 is 3.1203
the history score probability density function appears to have an unsampled region at the largest history scores:
please examine. see print table 161.

fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (4.227E+04)*(5.780E-02)**2 = (4.227E+04)*(3.341E-03) = 1.412E+02

1tally 25 nps = 10000000
tally type 5 particle flux at a point detector.
particle(s): photons
this tally is modified by standard dose function 1.

this tally is all multiplied by 2.25676E+07

detector located at x,y,z = 9.14300E+01 2.43840E+02 6.91975E+01
4.97439E-07 0.0024

detector located at x,y,z = 9.14300E+01 2.43840E+02 6.91975E+01
uncollided photon flux
1.65726E-07 0.0010

detector score diagnostics		cumulative	tally	cumulative
	fraction of	per	fraction of	
times average score	transmissions	transmissions	history	total tally
1.00000E-01	2719757	0.28347	1.67735E-09	0.02719

1.00000E+00	4570832	0.75986	1.01499E-08	0.19169
2.00000E+00	976550	0.86165	8.59433E-09	0.33098
5.00000E+00	959927	0.96169	1.86191E-08	0.63274
1.00000E+01	304493	0.99343	1.27090E-08	0.83871
1.00000E+02	55103	0.99917	6.01146E-09	0.93614
1.00000E+03	2382	0.99942	3.38705E-09	0.99104
1.00000E+38	61	0.99943	5.47142E-10	0.99990
before dd roulette	5491	1.00000	5.86511E-12	1.00000

average tally per history = 6.17012E-08 largest score = 1.98339E-04
(largest score)/(average tally) = 3.21450E+03 nps of largest score = 7980777

score contributions by cell

	cell	misses	hits	tally per history	weight per hit
1	10	47440116	8160282	3.45288E-08	4.23132E-08
2	50	5631580	772760	1.10836E-08	1.43429E-07
3	11	4742843	661554	1.60889E-08	2.43198E-07
	total	57814539	9594596	6.17012E-08	6.43083E-08

score misses

russian roulette on pd	0
psc=0.	855422
russian roulette in transmission	53023942
underflow in transmission	3935175
hit a zero-importance cell	0
energy cutoff	0

===

results of 10 statistical checks for the estimated answer for the tally fluctuation chart (tfc) bin of tally 25

tfc bin behavior	--mean-- behavior	---------relative error--------- value	decrease	decrease rate	----variance of the variance---- value	decrease	decrease rate	--figure of merit-- value	behavior	-pdf- slope
desired	random	<0.05	yes	1/sqrt(nps)	<0.10	yes	1/nps	constant	random	>3.00
observed	random	0.00	yes	yes	0.00	yes	yes	constant	random	4.00
passed?	yes	yes	yes	yes	yes	yes	yes	yes	yes	yes

===

this tally meets the statistical criteria used to form confidence intervals: check the tally fluctuation chart to verify.
the results in other bins associated with this tally may not meet these statistical criteria.

----- estimated confidence intervals: -----

estimated asymmetric confidence interval(1,2,3 sigma): 4.9627E-07 to 4.9866E-07; 4.9508E-07 to 4.9986E-07; 4.9388E-07 to 5.0105E-07
estimated symmetric confidence interval(1,2,3 sigma): 4.9625E-07 to 4.9863E-07; 4.9505E-07 to 4.9983E-07; 4.9386E-07 to 5.0102E-07

1analysis of the results in the tally fluctuation chart bin (tfc) for tally 25 with nps = 10000000 print table 160

normed average tally per history = 4.97439E-07 unnormed average tally per history = 4.97439E-07
estimated tally relative error = 0.0024 estimated variance of the variance = 0.0047
relative error from zero tallies = 0.0004 relative error from nonzero scores = 0.0024

number of nonzero history tallies = 4079202 efficiency for the nonzero tallies = 0.4079
history number of largest tally = 7980777 largest unnormalized history tally = 2.28189E-03
(largest tally)/(average tally) = 4.58726E+03 (largest tally)/(avg nonzero tally)= 1.87124E+03

(confidence interval shift)/mean = 0.0001 shifted confidence interval center = 4.97467E-07 if the largest history score sampled so far were to occur on the next history, the tfc bin quantities would change as follows:

estimated quantities	value at nps	value at nps+1	value(nps+1)/value(nps)-1.
mean	4.97439E-07	4.97668E-07	0.000459
relative error	2.40083E-03	2.44312E-03	0.017616
variance of the variance	4.71546E-03	5.62879E-03	0.193689
shifted center	4.97467E-07	4.97470E-07	0.000006
figure of merit	7.33269E+02	7.08102E+02	-0.034322

the estimated inverse power slope of the 200 largest tallies starting at 2.97193E-04 is 4.0044
the large score tail of the empirical history score probability density function appears to have no unsampled regions.

fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (4.227E+04)*(1.317E-01)**2 = (4.227E+04)*(1.735E-02) = 7.333E+02

1status of the statistical checks used to form confidence intervals for the mean for each tally bin

tally result of statistical checks for the tfc bin (the first check not passed is listed) and error magnitude check for all bins

15 passed the 10 statistical checks for the tally fluctuation chart bin result
passed all bin error check: 2 tally bins all have relative errors less than 0.05 with no zero bins

25 passed the 10 statistical checks for the tally fluctuation chart bin result
passed all bin error check: 2 tally bins all have relative errors less than 0.05 with no zero bins

the 10 statistical checks are only for the tally fluctuation chart bin and do not apply to other tally bins.

1tally fluctuation charts

	tally 15					tally 25				
nps	mean	error	vov	slope	fom	mean	error	vov	slope	fom
512000	2.4094E-05	0.0352	0.5579	2.2	83	4.9978E-07	0.0102	0.1024	3.3	988

1024000	2.4078E-05	0.0213	0.2766	2.2	110	4.9771E-07	0.0075	0.0438	3.2	876
1536000	2.3929E-05	0.0160	0.1849	2.3	136	4.9688E-07	0.0059	0.0257	3.1	989
2048000	2.3935E-05	0.0128	0.1420	2.3	153	4.9847E-07	0.0056	0.0281	3.2	796
2560000	2.3867E-05	0.0111	0.1081	2.4	150	4.9707E-07	0.0049	0.0214	3.8	776
3072000	2.3783E-05	0.0100	0.0890	2.6	160	4.9589E-07	0.0044	0.0171	3.9	832
3584000	2.3846E-05	0.0094	0.0676	3.0	161	4.9585E-07	0.0040	0.0136	4.0	880
4096000	2.3883E-05	0.0088	0.0556	3.0	161	4.9676E-07	0.0037	0.0110	3.5	899
4608000	2.3874E-05	0.0082	0.0469	2.9	161	4.9657E-07	0.0035	0.0098	3.5	870
5120000	2.4010E-05	0.0081	0.0377	2.8	153	4.9693E-07	0.0034	0.0087	3.6	879
5632000	2.3975E-05	0.0076	0.0338	2.6	161	4.9764E-07	0.0032	0.0075	3.9	888
6144000	2.3969E-05	0.0072	0.0300	2.7	157	4.9766E-07	0.0031	0.0069	3.4	824
6656000	2.3965E-05	0.0068	0.0273	2.7	150	4.9714E-07	0.0030	0.0063	3.1	784
7168000	2.3940E-05	0.0065	0.0253	2.7	154	4.9817E-07	0.0029	0.0056	3.3	770
7680000	2.3942E-05	0.0063	0.0236	2.6	150	4.9818E-07	0.0028	0.0051	3.5	777
8192000	2.3982E-05	0.0061	0.0208	3.0	143	4.9807E-07	0.0027	0.0062	3.2	724
8704000	2.3969E-05	0.0059	0.0190	3.1	143	4.9799E-07	0.0026	0.0057	3.2	726
9216000	2.3925E-05	0.0056	0.0184	3.2	144	4.9759E-07	0.0025	0.0053	3.3	721
9728000	2.3939E-05	0.0056	0.0194	3.1	138	4.9753E-07	0.0024	0.0049	3.7	723
10000000	2.3924E-05	0.0055	0.0190	3.1	141	4.9744E-07	0.0024	0.0047	4.0	733

****

dump no. 2 on file runtpe nps = 10000000 coll = 73644179 ctm = 236.60 nrn =
1463531465

5 warning messages so far.

run terminated when 10000000 particle histories were done.

computer time = 236.73 minutes

mcnp version 6 05/08/13 12/15/15 16:50:13 probid = 12/15/15 16:46:08

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
3. Have the appropriate client review forms/checklists been completed? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation? No current ENERCON calculations exist that are similar to this calculation.	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☒	☐	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation? No ENERCON design inputs or outputs are affected by this calculation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis? This calculation applies to a remediation site. No work performed in this calculation is applicable to a licensing basis.	☐	☐	☒
17. If so, is this documented in the calculation?	☐	☐	☒
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☐	☒
19. If so, is this documented in the calculation?	☐	☐	☒
20. Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☐	☒
21. If so, is this documented in the calculation?	☐	☐	☒
22. Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS			
23. Are design inputs clearly identified?	☒	☐	☐
24. Are design inputs retrievable or have they been added as attachments?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
25. If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☒	☐	☐
26. Are design inputs clearly distinguished from assumptions?	☒	☐	☐
DESIGN INPUTS (Continued)			
27. Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☒	☐	☐
28. Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29. Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☐	☐	☒
30. If applicable, do design inputs adequately address actual plant conditions?	☐	☐	☒
31. Are input values reasonable and correctly applied?	☒	☐	☐
32. Are design input sources approved? The Cimarron design is currently at 60% completion.	☐	☒	☐
33. Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34. Were all applicable plant operating modes considered?	☐	☐	☒
ASSUMPTIONS			
35. Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36. Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37. Are any engineering judgments used?	☒	☐	☐
38. Are engineering judgments clearly identified as such? No engineering judgments were applied in this evaluation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
39. If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☐	☒
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use? Equations applied in this evaluation are in common use in the industry.	☐	☐	☒
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input? Results generated by calculations performed in this evaluation are not significantly affected by minor perturbations of variables.	☐	☐	☒
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☒	☐	☐
SOFTWARE/COMPUTER CODES (Continued)			
52. Are the codes properly identified along with source vendor, organization, and revision level?	☒	☐	☐
53. Is the computer code applicable for the analysis being performed?	☒	☐	☐
54. If applicable, does the computer model adequately consider actual plant conditions?	☒	☐	☐
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☒	☐	☐
56. Is the computer output clearly identified?	☒	☐	☐
57. Does the computer output clearly identify the appropriate units? The output units are not identified in the output document. Tallies have been modified through multipliers and dose response functions. This process has been adequately documented within this calculation.	☐	☒	☐
58. Are the computer outputs reasonable when compared to the inputs and what was expected? Only basic functions and operations in Microsoft Excel 2013 were applied in this calculation.	☒	☐	☐
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results? While warning messages exist, they do not impact the results.	☒	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified? No acceptance criteria required for this evaluation.	☐	☐	☒
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☐	☐	☒
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☐	☐	☒
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations? No ENERCON calculations are affected by this evaluation.	☐	☐	☒
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☒	☐	☐

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator:

Caleb Trainor [signature] 12/21/2015

Print Name and Sign | Date