environmental
properties management, LLC

January 14, 2020

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
Evaluating the Need to License Tc-99

Dear Sirs:

Solely as Trustee for the Cimarron Environmental Response Trust (CERT), Environmental Properties Management LLC (EPM) submits herein an evaluation of potential exposure to technetium-99 (Tc-99) in groundwater and water treatment media and wastes at the Cimarron Site.

In 1996, the licensee performed an investigation to identify the source of elevated beta concentrations in groundwater. Tc-99 was identified as a contaminant in groundwater as a result of that investigation. After discovering that the Tc-99 was a contaminant in the uranium hexafluoride feedstock received at the facility, the US Nuclear Regulatory Commission (NRC) and the licensee deliberated the need to specifically list Tc-99 in the license.

In a letter dated April 22, 1997, the NRC stated, "… in accordance with Section 4.3 of Regulatory Guide 10.3, "Guide for the Preparation of Applications for Special Nuclear Material Licenses of Less Than Critical Mass Quantities," NRC staff has determined that radiological contaminants need not be specifically listed as an authorized material on licenses, unless they occur in sufficient quantities to pose unique or significant radiation hazards to workers or the public." Based on the presence of Tc-99 in groundwater, it was determined that Tc-99 did not need to be specifically licensed at that time.

The process of groundwater extraction and treatment presents potential exposure pathways not applicable to Tc-99 existing in groundwater below the surface. Groundwater containing concentrations of Tc-99 that are below primary drinking water standards will be extracted from the subsurface and may be present in detectable concentrations in several in-process materials. Those materials which may contain Tc-99 are:

1. Sediment that is greater than 10 microns in diameter will be filtered out of the groundwater prior to treatment. The concentration of Tc-99 that is sorbed onto the sediment is a function of the distribution coefficient and the concentration of Tc-99 in the influent. Because literature values for the distribution coefficient of Tc-99 are much less than 1, the concentration of Tc-99 in the sediment is expected to be a small fraction of the concentration of Tc-99 in the groundwater. Most of the Tc-99 present in the groundwater will flow through the filters to vessels containing ion exchange resin.
2. Ion exchange resin is expected to remove most, if not all, of the Tc-99 from the influent groundwater. For the purpose of this evaluation, it was conservatively assumed that ***all*** the Tc-99 in the influent is captured by the ion exchange resin.
3. Bioreactor solids (referred to herein as biomass) will be generated in the biodenitrification process. Most, if not all, of the Tc-99 in the influent will be removed by the ion exchange treatment system. For the purpose of this evaluation, it was conservatively assumed that ***all*** the Tc-99 in the influent passes through the ion exchange system and becomes absorbed onto the biomass.

Although all three of these waste streams will be moist, and the potential for these materials to become airborne will be negligible, the following exposure pathways were assumed to be viable pathways, and were evaluated for this submittal:

1. External exposure to the whole body
2. External exposure to the skin
3. Inhalation
4. Oral ingestion

Conservative assumptions were applied in the evaluation of each exposure pathway. Attachment 1 to this letter is a paper entitled, "Licensing Considerations for the Presence of Tc-99". The calculations presented therein were performed by Enercon Services, Inc. (Enercon) at the request of EPM.

EPM maintains that the calculations provided in this paper, which employs extremely conservative assumptions, demonstrates that Tc-99 in groundwater and treatment media and wastes presents "no unique or significant radiation hazards to workers or the public". Consequently, Tc-99 need not be specifically licensed.

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EPM requests NRC review of the attached calculations and concurrence that Tc-99 need not be specifically licensed. If you have questions or comments, please contact me at 405-641-5152 or at jlux@envpm.com.

Sincerely,

Jeff Lux

Jeff Lux, P.E.
Trustee Project Manager

Attachment

cc: Michael Broderick, Oklahoma Department of Environmental Quality (electronic copy only)
NRC Public Document Room (electronic copy only)

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ATTACHMENT 1
LICENSING CONSIDERATIONS FOR THE PRESENCE OF Tc-99

ENERCON *Excellence—Every project. Every day*	CALCULATION COVER SHEET	CALC NO.	EPM027-CALC-001
		REV.	0
		PAGE NO.	1 of 21

Title:	**Licensing Considerations for the Presence of Tc-99**	**Client:**	**Cimarron Environmental Response Trust**
		Project Identifier:	**EPM027**

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:

Revision Impact on Results:

Study Calculation ☐ **Final Calculation** ☒

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

(Print Name and Sign)

Originator: A. Joseph Nardi	A Joseph Nardi	**Date:**	**12/16/2019**
Design Reviewer: Jay Maisler	J Maisler	**Date:**	12/27/2019
Approver: Gerald Williams	Gerald Williams Digitally signed by Gerald Williams Date: 2020.01.07 04:56:50 -05'00'	**Date:**	

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		REV. 0
		PAGE NO. 2 of 21

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	12/16/2019	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	7	0

1.0 Purpose and Scope

The purpose of this calculation is to estimate the significance of Tc-99 to various routes of radiation exposure associated with materials that will be present during the operation of the Groundwater Treatment Facility at the Cimarron Site. The methodology used is to show that the relative significance of the radiation dose received from Tc-99 exposure is insignificant. The potential for radiation exposure from Tc-99 is based on: 1) calculating the contribution to a radiation exposure pathway for various materials in comparison to the Uranium plus Progeny contribution and 2) a conservative estimate of the annual dose attributable to the Tc-99 present in that pathway. Contribution from the Tc-99 that is less than 10% of the total dose or a small fraction of the applicable dose limit can be considered insignificant.

2.0 Summary of Results and Conclusions

In all cases evaluated, the contribution to the total radiation exposure from the Tc-99 that may be present is less than 1% of the total exposure, and the associated dose rate is insignificant in magnitude. It is therefore concluded that the presence of the Tc-99 on the site is not a significant contributor to the radiation exposure potential or the licensing considerations for the site in accordance with the guidance provided in Reference 3.10, Section 3.3.

3.0 References

3.1 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

3.2 EPA 402-R-93-081, External Exposure To Radionuclides In Air, Water, And Soil, Federal Guidance Report No. 12, September 1993

3.3 Burns & McDonnell Memorandum No. BMCD-GWREMED-TM004, Impact of Sediment, Technetium-99 and Bioreactor Sludge on Waste Generated by Cimarron Remediation Water Treatment Systems, Rev. D, September 10, 2019

3.4 EPM028-CALC-001, Potential Intake Calculation, Rev. 2

3.5 DOE-STD-1136-2004, Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities", December 2004

3.6 "A Review And Verification Of The Isotopic Distribution Of Enriched Uranium And The Impact On Decommissioning Considerations", A. J. Nardi, Tracy Chance and John F. Conant, Presented at the HPS 2007 Midyear Topical Meeting, Jan. 21-24, 2007.

3.7 EPM017-CALC-001, Dose Rate near Uranium Treatment Train, Rev 0, Dec. 21, 2015

3.8 Spreadsheet "VNS-EPM-004-CALC-D-001 RB-Mass Balance Excel.xlsx", Sheet "WATF Consumable Usage", Cell K7

3.9 The Health Physics and Radiological Health Handbook, Revised Edition edited by B. Shleien; 1992

3.10 NUREG-1757, Vol. 2 Revision 1, "Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria", September 2006

4.0 Assumptions

1) Tc-99 is present in groundwater only in the Western Alluvial Area (WAA). The U-235 enrichment of uranium in groundwater in this area was calculated to be 2.6% by weight (at a 95% confidence level).
2) The specific activity and isotopic distribution for 2.6% enriched uranium: (Reference 3.6)
   a) Specific Activity of 2.6% enrichment – 1.41E-06 Ci/g
   b) Isotopic Activity Distribution
      - U-234 – 72.8%
      - U-235 – 4.0%
      - U-238 – 23.2%
3) The enriched uranium is considered to have returned to secular equilibrium with the progeny appropriate for chemically separated uranium. The following radionuclides were included in the analysis:

Tc-99	U-234	U-235	Th-231	U-238	Th-234	Pa-234m

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5.0 Design Inputs

1) The ALI used to compare the inhalation and oral ingestion pathways for internal dose were based on Reference 3.1. For this evaluation the most conservative ALI was utilized (Stochastic or Non-stochastic). These values are:

Radionuclide	Inhalation Annual Limit of Intake (ALI) (See Note 1)	Oral Ingestion Annual Limit of Intake (ALI) (See Note 1)
	(microCi)	(microCi)
Tc-99	7.E+02	4.E+03
U-234	4.E-02	1.E+01
U-235	4.E-02	1.E+01
Th-231	6.E+03	4.E+03
U-238	4.E-02	1.E+01
Th-234	2.E+02	3.E+02
Pa-234m	7.E+03	2.E+03

Note 1: 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. This reference only provides a listing for Pa-234 which has been used for the values for Pa-234m.

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2) The Dose Conversion Factors used to compare the Effective and Skin Dose rates were based on Reference 3.2. These values are:

Radionuclide	Dose Conversion Coefficient (See Note 1)	Dose Conversion Coefficient (See Note 2)
	$Sv/(Bq\text{-}s\text{-}m^{-3})$	(mrem/y)/(μCi/g)
Effective Dose		
Tc-99	6.72E-22	1.26E+02
U-234	2.15E-21	4.02E+02
U-235	3.86E-18	7.21E+05
Th-231	1.95E-19	3.64E+04
U-238	5.52E-22	1.03E+02
Th-234	1.29E-19	2.41E+04
Pa-234m	4.80E-19	8.97E+04
Skin Dose		
Tc-99	9.09E-22	1.70E+02
U-234	5.99E-21	1.12E+03
U-235	4.40E-18	8.22E+05
Th-231	2.56E-19	4.78E+04
U-238	3.55E-21	6.63E+02
Th-234	1.50E-19	2.80E+04
Pa-234m	8.27E-18	1.54E+06
Note 1: Dose Conversion Coefficients - Exposure to soil contaminated to an infinite depth (FGR Report Number 12, Table III.7)-(Ref.3.2)		
Note 2: The SI value is multiplied by 1.868E23 to obtain these values.		

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6.0 Methodology

6.1 Approaches

Two approaches are considered in this evaluation.

1) The first approach is to consider the percent contribution to dose of the Tc-99 in comparison to the percent contribution of the uranium (including its progeny). This analysis demonstrates that the Tc-99 is insignificant in comparison to the uranium component. These calculations are based on the composition of three materials considered in the analysis. The analysis does not consider a specific exposure scenario that incorporates realistic geometry or release fractions and therefore is not intended to estimate the absolute dose rate associated with the materials considered. The results are presented in terms of the percent contribution of the Tc-99 in comparison to the combined contribution of all the above-mentioned nuclides for four occupational exposures; external exposure to the whole body, external exposure to the skin, inhalation and ingestion. These results are presented in Section 7.0.
2) The second approach translates the results of the first approach to dose rate values in a conservative manner. For the effective dose and the skin dose, the analysis from the first approach provides conservative annual doses for the three materials. Those annual doses are not based on realistic exposure conditions because they conservatively represent the annual dose to an individual above an unshielded infinite plane of the material which is modeled as contaminated soil rather than the actual material being considered in the stored or handled condition.
   For the inhalation and oral ingestion dose calculations, the assumption is made that the individual has had an annual intake of one ALI (equivalent to an occupational dose of 5,000 mrem) of the resin mixture. This is a bounding analysis without regard to consideration of how that intake might have occurred. Thus, this approach represents a bounding condition for the occupational annual dose from the Tc-99. These results are presented in Section 8.0.

For comparison, two other evaluations are discussed that are based on more realistic models for radiation exposure. One model is the dose rate near a loaded resin column (Ref. 3.7) and the second is the potential intake for the quantity of a radionuclide handled in a year (Ref. 3.4). These comparisons are presented in Section 9.0.

Considerations regarding the dose to the public is discussed in Section 10.0.

6.2 Materials Considered

Three materials were considered for this evaluation:

1. Spent resin material
2. Biomass material
3. Filtered solids material

6.2.1 Spent Resin Material

The isotopic concentration of this material is based on:

- All the Tc-99 in the groundwater that passes through the column is captured by the resin material,
- The uranium loaded on the resin material is based on the uranium loading for the initial resin column for treatment of groundwater from the WATF area. (Reference 3.8)

6.2.2 Biomass Material

The isotopic concentration of this material is based on:

- None of the Tc-99 is captured by the resin and all of it is captured by the biomass material,
- The uranium concentration of this material was taken from Reference 3.3.

6.2.3 Filtered Solids Material

The isotopic concentration of the Tc-99 and the uranium in this material is based on Reference 3.3.

6.3 Radiation Exposure Pathways Considered

The following exposure pathways are considered to compare the contribution of the Tc-99 and Uranium for that pathway:

- External exposure to the whole body
- External exposure to the skin
- Inhalation pathway
- Oral Ingestion pathway

6.4 Comparison of Tc-99 and Uranium Pathways

To compare the effective dose rate and skin dose rate for each material, the dose rate for each radionuclide was calculated (Dose Conversion Coefficient X Radionuclide Concentration). The % contribution to the total dose rate for each radionuclide was then calculated.

To compare the inhalation and oral ingestion pathways for each material, the ratio of the concentration to the ALI was calculated for each material. This is a relative value of the importance of each radionuclide to the Sum-of-Fractions of the total ALI and is a direct measure of the importance of that radionuclide for that pathway for that material. Radionuclides that provide a total contribution of less than 10% to the total Sum-of-Fractions are considered insignificant.

7.0 Calculation of Tc-99 Dose Contribution

7.1 Radionuclide concentrations in the Resin Material

Tc-99 - Assuming all Tc-99 is captured by resin		
466	pCi/L	Initial (and maximum) Tc-99 Groundwater concentration (Ref. 3.3, Sec 2.2.2)
750	kg	Mass of Spent resin in Column (DP Rev. 1, Section 13.1.1)
130	gpm	Flow rate to Column (Ref. 3.3, Table 1) (See Note 1)
492.05	L/min	Flow rate to Column
129600	min	Time until resin column is spent (Based on 90 day cycle for column)
2.97E+10	pCi	Total pCi Tc-99 captured by column
3.96E+04	pCi/g	Tc-99 activity per gram of resin
Uranium		
3.09E-04	Ci U-235	U-235 activity in starting column for WATF resin (Ref. 3.8)
412	pCi/g	U-235 concentration of U-235 in WATF resin
2.6%	percent	Weight percent of U-235 in Uranium (Ref. 3.3, Sec 4.1)
72.8%	percent	Activity % of U-234 in Uranium (Ref. 3.6)
4.0%	percent	Activity % of U-235 in Uranium (Ref. 3.6)
23.2%	percent	Activity % of U-238 in Uranium (Ref. 3.6)
7.50E+03	pCi/g	U-234 activity per gram of resin
4.12E+02	pCi/g	U-235 activity per gram of resin
4.12E+02	pCi/g	Th-231 activity per gram of resin
2.39E+03	pCi/g	U-238 activity per gram of resin
2.39E+03	pCi/g	Th-234 activity per gram of resin
2.39E+03	pCi/g	Pa-234 activity per gram of resin\

Note 1 : The Table list 250 gpm as the flow from the WATF field. Since this flow feeds two treatment trains, a value of 130 gpm was used as the feed to one train.

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7.2 Radionuclide concentrations in the Biomass Material

Tc-99 - Assuming all Tc-99 is captured by Biomass and none by Resin		
466	pCi/L	Initial (and maximum) Tc-99 Groundwater concentration (Ref. 3.3, Sec 2.2.2)
2895	lbs/day	Mass of Spent Biomass at 150 ppm Nitrate (Ref. 303, Table 7)
250	gpm	Flow rate from both columns Column (Ref. 3.3, Table 1)
946.25	L/min	Flow rate to Column
129600	min	Time until resin column is spent (Based on 90 day cycle for column)
5.71E+10	pCi	Total pCi Tc-99 passed by column
1.34E-01	pCi/g	Tc-99 activity per gram of Biomass
Uranium - Assuming Uranium Concentration in Biomass per BMCD-GWREMED-TM004, Table 7 @ 150 ppm Nitrate (Ref. 3.3)		
0.113	pCi/g	Uranium concentration in Biomass (Ref. 3.3, Table 7)
2.6%	percent	Weight percent of U-235 in Uranium (Ref. 3.3, Sec 4.1)
72.8%	percent	Activity % of U-234 in Uranium (Ref. 3.6)
4.0%	percent	Activity % of U-235 in Uranium (Ref. 3.6)
23.2%	percent	Activity % of U-238 in Uranium (Ref. 3.6)
8.23E-02	pCi/g	U-234 activity per gram of resin
4.52E-03	pCi/g	U-235 activity per gram of resin
4.52E-03	pCi/g	Th-231 activity per gram of resin
2.62E-02	pCi/g	U-238 activity per gram of resin
2.62E-02	pCi/g	Th-234 activity per gram of resin
2.62E-02	pCi/g	Pa-234 activity per gram of resin\

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7.3 Radionuclide concentrations in the Filtered Solids Material

Tc-99 - Filtered Solids From WATF per BMCD-GWREMED-TM004, Table 3		
0.05	pCi/g	Initial (and Maximum) Tc-99 in Filtered Solids (Ref. 3.3, Section 2.2.2)
Uranium - Filtered Solids from WATF per BMCD-GWREMED-TM004, Table 2		
2.0	pCi/g	Uranium concentration in Filtered Solids (Ref.3.3, Sec. 2.2.1)
2.6%	percent	Weight percent of U-235 in Uranium (Ref. 3.3, Sec 4.1)
72.8%	percent	Activity % of U-234 in Uranium (Ref. 3.6)
4.0%	percent	Activity % of U-235 in Uranium (Ref. 3.6)
23.2%	percent	Activity % of U-238 in Uranium (Ref. 3.6)
1.46E+00	pCi/g	U-234 activity per gram of resin
8.00E-02	pCi/g	U-235 activity per gram of resin
8.00E-02	pCi/g	Th-231 activity per gram of resin
4.64E-01	pCi/g	U-238 activity per gram of resin
4.64E-01	pCi/g	Th-234 activity per gram of resin
4.64E-01	pCi/g	Pa-234 activity per gram of resin\

7.4 Summary of Calculated Radionuclide Activity Concentrations

Activity Concentrations for Materials

Radionuclide	Resin Activity	Biomass	Filtered Solids
	pCi/g	pCi/g	pCi/g
Tc-99	3.96E+04	1.34E-01	5.00E-02
U-234	7.50E+03	8.23E-02	1.46E+00
U-235	4.12E+02	4.52E-03	8.00E-02
Th-231	4.12E+02	4.52E-03	8.00E-02
U-238	2.39E+03	2.62E-02	4.64E-01
Th-234	2.39E+03	2.62E-02	4.64E-01
Pa-234m	2.39E+03	2.62E-02	4.64E-01

7.5 Comparison of External Dose Rates for Resin Material

External Dose Rate Calculations for Resin Material					
Effective Dose from Resin Material					
Radionuclide	Dose Conversion Coefficient	Radionuclide Concentration in Resin	Percent Distribution of Activity in Resin	Contribution to Dose Rate	Percent of Contribution to Dose Rate
	(mrem/y)/(μCi/g)	pCi/g	%	mrem/yr	%
Tc-99	1.26E+02	3.96E+04	71.9%	4.97E+00	0.84%
U-234	4.02E+02	7.50E+03	13.6%	3.01E+00	0.51%
U-235	7.21E+05	4.12E+02	0.7%	2.97E+02	50.2%
Th-231	3.64E+04	4.12E+02	0.7%	1.50E+01	2.5%
U-238	1.03E+02	2.39E+03	4.3%	2.46E-01	0.042%
Th-234	2.41E+04	2.39E+03	4.3%	5.76E+01	9.7%
Pa-234m	8.97E+04	2.39E+03	4.3%	2.14E+02	36.2%
	Total	5.51E+04	100.0%	5.92E+02	100.0%
Contribution to Effective Dose Rate from Uranium Isotopes plus Progeny					99.16%
	Contribution to Effective Dose Rate from Tc-99				0.84%

Skin Dose from Resin Material					
Radionuclide	Dose Conversion Coefficient	Radionuclide Concentration in Resin	Percent Distribution of Activity in Resin	Contribution to Dose Rate	Percent of Contribution to Dose Rate
	(mrem/y)/(μCi/g)	pCi/g	%	mrem/yr	%
Tc-99	1.70E+02	3.96E+04	71.9%	6.73E+00	0.163%
U-234	1.12E+03	7.50E+03	13.6%	8.39E+00	0.20%
U-235	8.22E+05	4.12E+02	0.7%	3.39E+02	8.2%
Th-231	4.78E+04	4.12E+02	0.7%	1.97E+01	0.48%
U-238	6.63E+02	2.39E+03	4.3%	1.58E+00	0.038%
Th-234	2.80E+04	2.39E+03	4.3%	6.70E+01	1.6%
Pa-234m	1.54E+06	2.39E+03	4.3%	3.69E+03	89.3%
	Total	5.51E+04	100.0%	4.13E+03	100.0%
Contribution to Skin Dose Rate from Uranium Isotopes plus Progeny					99.84%
	Contribution to Skin Dose Rate from Tc-99				0.163%
Note 1: Dose Conversion Coefficients - Exposure to soil contaminated to an infinite depth (FGR Report Number 12, Table III.7)-(Ref.3.2). See Section 5.0					

7.6 Comparison of External Dose Rates for Biomass Material

External Dose Rate Calculations for Biomass Material					
Effective Dose from Biomass Material					
Radionuclide	Dose Conversion Coefficient	Radionuclide Concentration in Biomass	Percent Distribution of Activity in Biomass	Contribution to Dose Rate	Percent of Contribution to Dose Rate
	(mrem/y)/ (μCi/g)	pCi/g	%	mrem/yr	%
Tc-99	1.26E+02	1.34E-01	44.1%	1.68E-05	0.26%
U-234	4.02E+02	8.23E-02	27.0%	3.30E-05	0.51%
U-235	7.21E+05	4.52E-03	1.5%	3.26E-03	50.5%
Th-231	3.64E+04	4.52E-03	1.5%	1.65E-04	2.5%
U-238	1.03E+02	2.62E-02	8.6%	2.70E-06	0.042%
Th-234	2.41E+04	2.62E-02	8.6%	6.32E-04	9.8%
Pa-234m	8.97E+04	2.62E-02	8.6%	2.35E-03	36.4%
	Total	3.04E-01	100.0%	6.46E-03	100.0%
Contribution to Effective Dose Rate from Uranium Isotopes plus Progeny					99.7%
Contribution to Effective Dose Rate from Tc-99					0.26%

Skin Dose from Biomass Material					
Radionuclide	Dose Conversion Coefficient	Radionuclide Concentration in Biomass	Percent Distribution of Activity in Biomass	Contribution to Dose Rate	Percent of Contribution to Dose Rate
	(mrem/y)/ (μCi/g)	pCi/g	%	mrem/yr	%
Tc-99	1.70E+02	1.34E-01	44.1%	2.28E-05	0.050%
U-234	1.12E+03	8.23E-02	27.0%	9.20E-05	0.20%
U-235	8.22E+05	4.52E-03	1.5%	3.72E-03	8.2%
Th-231	4.78E+04	4.52E-03	1.5%	2.16E-04	0.5%
U-238	6.63E+02	2.62E-02	8.6%	1.74E-05	0.038%
Th-234	2.80E+04	2.62E-02	8.6%	7.35E-04	1.6%
Pa-234m	1.54E+06	2.62E-02	8.6%	4.05E-02	89.4%
	Total	3.04E-01	100.0%	4.53E-02	100.0%
Contribution to Skin Dose Rate from Uranium Isotopes plus Progeny					99.95%
Contribution to Skin Dose Rate from Tc-99					0.050%

Note 1: Dose Conversion Coefficients - Exposure to soil contaminated to an infinite depth (FGR Report Number 12, Table III.7)-(Ref.3.2). See Section 5.0

7.7 Comparison of External Dose Rates for Filtered Solids Material

External Dose Rate Calculations for Filtered Solids Material					
Effective Dose from Filtered Solids Material					
Radionuclide	Dose Conversion Coefficient	Radionuclide Concentration in Filtered Solids	Percent Distribution of Activity in Filtered Solids	Contribution to Dose Rate	Percent of Contribution to Dose Rate
	(mrem/y)/(µCi/g)	pCi/g	%	mrem/yr	%
Tc-99	1.26E+02	5.00E-02	1.6%	6.28E-06	0.006%
U-234	4.02E+02	1.46E+00	47.6%	5.85E-04	0.5%
U-235	7.21E+05	8.00E-02	2.6%	5.77E-02	50.6%
Th-231	3.64E+04	8.00E-02	2.6%	2.91E-03	2.6%
U-238	1.03E+02	4.64E-01	15.2%	4.78E-05	0.042%
Th-234	2.41E+04	4.64E-01	15.2%	1.12E-02	9.8%
Pa-234m	8.97E+04	4.64E-01	15.2%	4.16E-02	36.5%
	Total	3.06E+00	100.0%	1.14E-01	100.0%
Contribution to Effective Dose Rate from Uranium Isotopes plus Progeny					99.994%
	Contribution to Effective Dose Rate from Tc-99				0.006%
Skin Dose from Filtered Solids Material					
Radionuclide	Dose Conversion Coefficient	Radionuclide Concentration in Filtered Solids	Percent Distribution of Activity in Filtered Solids	Contribution to Dose Rate	Percent of Contribution to Dose Rate
	(mrem/y)/(µCi/g)	pCi/g	%	mrem/yr	%
Tc-99	1.70E+02	5.00E-02	1.6%	8.49E-06	0.001%
U-234	1.12E+03	1.46E+00	47.6%	1.63E-03	0.2%
U-235	8.22E+05	8.00E-02	2.6%	6.58E-02	8.2%
Th-231	4.78E+04	8.00E-02	2.6%	3.83E-03	0.5%
U-238	6.63E+02	4.64E-01	15.2%	3.08E-04	0.038%
Th-234	2.80E+04	4.64E-01	15.2%	1.30E-02	1.6%
Pa-234m	1.54E+06	4.64E-01	15.2%	7.17E-01	89.5%
	Total	3.06E+00	100.0%	8.01E-01	100.0%
Contribution to Skin Dose Rate from Uranium Isotopes plus Progeny					99.999%
	Contribution to Skin Dose Rate from Tc-99				0.001%
Note 1: Dose Conversion Coefficients - Exposure to soil contaminated to an infinite depth (FGR Report Number 12, Table III.7)-(Ref.3.2). See Section 5.0					

7.8 Calculation of Percent Contribution to Dose for Inhalation Pathway

Contribution to Dose Calculation for Airborne Activity of Resin Material				
Radionuclide	Annual Limit of Intake (ALI) (See Note 1)	Activity Concentration	Relative Ratio	Percent Contribution to Dose
	(microCi)	(microCi/g)	(Ratio)	(Percent)
(a)	(b)	(c)	(d) = (c)/(b)	(e)=[(d)/Total]*100
Tc-99	7E+02	3.96E-02	5.66E-05	0.022%
U-234	4E-02	7.50E-03	1.87E-01	72.78%
U-235	4E-02	4.12E-04	1.03E-02	4.00%
Th-231	6E+03	4.12E-04	6.87E-08	0.00003%
U-238	4E-02	2.39E-03	5.97E-02	23.19%
Th-234	2E+02	2.39E-03	1.19E-05	0.005%
Pa-234m	7E+03	2.39E-03	3.41E-07	0.0001%
		Total	2.58E-01	100%
Contribution to Airborne Dose from Uranium Isotopes + Progeny				99.98%
Contribution to Airborne Dose from Tc-99				0.02%

Contribution to Dose Calculation for Airborne Activity of Biomass Material				
Radionuclide	Annual Limit of Intake (ALI) (See Note 1)	Activity Concentration	Relative Ratio	Percent Contribution to Dose
	(microCi)	(microCi/g)	(Ratio)	(Percent)
(a)	(b)	(c)	(d) = (c)/(b)	(e)=[(d)/Total]*100
Tc-99	7E+02	1.34E-07	1.92E-10	0.007%
U-234	4E-02	8.23E-08	2.06E-06	72.79%
U-235	4E-02	4.52E-09	1.13E-07	4.00%
Th-231	6E+03	4.52E-09	7.53E-13	0.00003%
U-238	4E-02	2.62E-08	6.55E-07	23.20%
Th-234	2E+02	2.62E-08	1.31E-10	0.005%
Pa-234m	7E+03	2.62E-08	3.75E-12	0.0001%
		Total	2.83E-06	100%
Contribution to Airborne Dose from Uranium Isotopes + Progeny				99.993%
Contribution to Airborne Dose from Tc-99				0.007%

Sum of Fractions Calculation for Airborne Activity of Filtered Solids Material				
Radionuclide	Annual Limit of Intake (ALI) (See Note 1)	Activity Concentration	Relative Ratio	Percent Contribution to Dose
	(microCi)	(microCi/g)	(Ratio)	(Percent)
(a)	(b)	(c)	(d) = (c)/(b)	(e)=[(d)/Total]*100
Tc-99	7E+02	5.00E-08	7.14E-11	0.0001%
U-234	4E-02	1.46E-06	3.64E-05	72.80%
U-235	4E-02	8.00E-08	2.00E-06	4.00%
Th-231	6E+03	8.00E-08	1.33E-11	0.00003%
U-238	4E-02	4.64E-07	1.16E-05	23.20%
Th-234	2E+02	4.64E-07	2.32E-09	0.005%
Pa-234m	7E+03	4.64E-07	6.63E-11	0.0001%
		Total	5.00E-05	100%
Contribution to Airborne Dose from Uranium Isotopes + Progeny				99.9999%
Contribution to Airborne Dose from Tc-99				0.0001%

Note 1: 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 (Ref. 3.1)

7.9 Calculation of Percent Contribution to Dose for Oral Ingestion Pathway

Contribution to Dose Calculation for Oral Ingestion of Resin Material				
Radionuclide	Annual Limit of Intake (ALI) (See Note 1)	Activity Concentration	Relative Ratio	Percent Contribution to Dose
	(microCi)	(microCi/g)	(Ratio)	(Percent)
(a)	(b)	(c)	(d) = (c)/(b)	[(d)/Total]*100
Tc-99	4E+03	3.96E-02	9.91E-06	0.94%
U-234	1E+01	7.50E-03	7.50E-04	71.47%
U-235	1E+01	4.12E-04	4.12E-05	3.93%
Th-231	4E+03	4.12E-04	1.03E-07	0.0098%
U-238	1E+01	2.39E-03	2.39E-04	22.78%
Th-234	3E+02	2.39E-03	7.97E-06	0.76%
Pa-234m	2E+03	2.39E-03	1.19E-06	0.114%
		Total	1.05E-03	100%
Contribution to Oral Ingestion Dose from Uranium Isotopes + Progeny				99.1%
Contribution to Oral Ingestion Dose from Tc-99				0.94%

Contribution to Dose Calculation for Oral Ingestion of Biomass Material				
Radionuclide	Annual Limit of Intake (ALI) (See Note 1)	Activity Concentration	Relative Ratio	Percent Contribution to Dose
	(microCi)	(microCi/g)	(Ratio)	(Percent)
(a)	(b)	(c)	(d) = (c)/(b)	[(d)/Total]*100
Tc-99	4E+03	1.34E-07	3.35E-11	0.29%
U-234	1E+01	8.23E-08	8.23E-09	71.94%
U-235	1E+01	4.52E-09	4.52E-10	3.95%
Th-231	4E+03	4.52E-09	1.13E-12	0.00988%
U-238	1E+01	2.62E-08	2.62E-09	22.93%
Th-234	3E+02	2.62E-08	8.74E-11	0.76%
Pa-234m	2E+03	2.62E-08	1.31E-11	0.1146%
		Total	1.14E-08	100%
Contribution to Oral Ingestion Dose from Uranium Isotopes + Progeny				99.7%
Contribution to Oral Ingestion Dose from Tc-99				0.29%

Contribution to Dose Calculation for Oral Ingestion of Filtered Solids Material				
Radionuclide	Annual Limit of Intake (ALI) (See Note 1)	Activity Concentration	Relative Ratio	Percent Contribution to Dose
	(microCi)	(microCi/g)	(Ratio)	(Percent)
(a)	(b)	(c)	(d) = (c)/(b)	[(d)/Total]*100
Tc-99	4E+03	5.00E-08	1.25E-11	0.006%
U-234	1E+01	1.46E-06	1.46E-07	72.15%
U-235	1E+01	8.00E-08	8.00E-09	3.96%
Th-231	4E+03	8.00E-08	2.00E-11	0.0099%
U-238	1E+01	4.64E-07	4.64E-08	22.99%
Th-234	3E+02	4.64E-07	1.55E-09	0.77%
Pa-234m	2E+03	4.64E-07	2.32E-10	0.115%
		Total	2.02E-07	100%
Contribution to Oral Ingestion Dose from Uranium Isotopes + Progeny				99.994%
Contribution to Oral Ingestion Dose from Tc-99				0.006%

Note 1: 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 (Ref. 3.1)

7.10 Summary of Calculated Percent Contributions

Radiation Exposure Pathway	Material	Percent Contribution	
		Uranium + Progeny	Tc-99
Effective Dose	Resin	99.16%	0.84%
	Biomass	99.74%	0.26%
	Filtered Solids	99.994%	0.006%
Skin Dose	Resin	99.84%	0.163%
	Biomass	99.95%	0.050%
	Filtered Solids	99.999%	0.0011%
Inhalation Pathway	Resin	99.978%	0.022%
	Biomass	99.993%	0.0068%
	Filtered Solids	99.9999%	0.00014%
Oral Ingestion Pathway	Resin	99.06%	0.94%
	Biomass	99.71%	0.29%
	Filtered Solids	99.994%	0.0062%

These calculations demonstrate that the contribution of Tc-99 to the occupational dose rate, relative to the uranium + progeny dose rate, is not significant. In all cases the contribution of the Tc-99 is less than 1% and can therefore be ignored (Reference 3.10, Section 3.3). The external exposure calculations are conservative for Tc-99 because no credit is taken for the shielding that would be present in the form of protective clothing and containers.

7.11 Conclusions

The summary table provided in Section 7.10 demonstrates that, relative to the dose rate from the uranium + progeny, the Tc-99 represents an insignificant contribution. In accordance with guidance provided in Reference 3.10, Section 3.3, the Tc-99 meets the criterion for an insignificant radionuclide. Reference 3.10 states that for radionuclides or pathways for which the sum of the dose contributions from all radionuclides and pathways are no greater than 10 % of the dose criteria are considered insignificant. Such radionuclides or pathways are insignificant contributors to dose and may be eliminated from further detailed consideration.

8.0 Calculation of Conservative Annual Dose for Tc-99

The following provides a more detailed analysis for each dose pathway that demonstrates that not only is the Tc-99 insignificant in comparison to the uranium + progeny contribution but that the dose considerations are also insignificant.

8.1 General Conservatism in Analysis

Assumptions are made in this analysis of the starting activity concentrations to be used for each of the three materials selected for analysis (Sections 7.1 through 7.3). In general, the assumptions used to generate these material compositions are intended to maximize the Tc-99 concentrations. Even if the uranium concentrations are reduced, the impact on the conclusion that the Tc-99 is an insignificant contributor will not change.

8.2 External and Skin Dose Pathways

The contribution of Tc-99 to the external and skin dose pathways are provided in Sections 7.5 through 7.7. The highest external and skin dose rates are for the resin materials. For the resin material (Section 7.5) the contribution of the Tc-99 to the annual dose rate is 5 mrem/yr for the external dose pathway and 7 mrem/yr for the skin dose pathway. Therefore, the conservatively estimated dose rate contribution from the Tc-99 is insignificant for both the external and skin pathways.

8.3 Inhalation and Oral Ingestion Dose Pathways

The contribution of Tc-99 to the inhalation and oral ingestion dose pathways are provided in Sections 7.8 and 7.9. The highest inhalation and oral ingestion contributions are for the resin materials. For the resin material the contribution of the Tc-99 to the ALI is 0.008% and for ingestion (Section 7.8) and 0.36% for oral ingestion (Section 7.9). This is equivalent to 0.4 mrem/yr for the inhalation pathway and 18 mrem/yr for the oral ingestion pathway assuming the individual had an intake of one ALI, equivalent to a dose of 5,000 mrem/yr. This assumption is without consideration for the realistic potentials for intake. Even with this conservative assumption, the estimated contributions from the Tc-99 is insignificant for both the inhalation and oral ingestion pathways.

9.0 Comparison with other Evaluations

Two results of this evaluation can be compared with two other evaluations that have been separately conducted.

9.1 External Dose Rate for Resin Bed Columns

Reference 3.7 provides the calculation of the external dose rate at 1 foot from a resin column that has been loaded with enriched uranium. The dose rate is given as 0.024 mrem/hr which equates to 210 mrem/yr. Section 7.5 gives the dose rate as 1,540 mrem/yr. The dose conversion factors (Sec. 5.0) of this evaluation are based the dose rate of an infinite plane of contaminated soil without intervening shielding. Reference 3.7 is based on the physical dimensions of the resin columns including the shielding provided by the steel walls of the resin vessels. Thus, it can be concluded that the results presented in this evaluation are consistent with the separate evaluation.

9.2 Calculation of Potential Intake

Reference 3.4 provides an evaluation of resin material of the potential airborne intake of Uranium and Tc-99 using the methodology of NUREG-1400, "Air Sampling in the Workplace, September 1993". That evaluation concludes that the potential intake of Tc-99 is 0.22% ALI. Section 8.3 calculates that under the conservative assumptions used the annual intake of Tc-99 would be 0.008% ALI. Thus, it can be concluded that the results presented in this evaluation are reasonably consistent with the separate evaluation given the different approaches used.

10.0 Consideration of Exposure to the Public from the Presence of Tc-99

Tc-99 emits a low energy beta particle that would be fully shielded by the containers used when transporting any of the waste materials. The thickness of steel or wood necessary to stop the beta emissions from Tc-99 is less than 1/8th inch and the range of the Tc-99 beta in air is approximately 4 feet. (Ref. 3.9, page 51) Thus, there is no credible exposure to the public from the transportation of waste material due to the Tc-99 present in the waste.

Airborne effluents of the three operational materials considered in Section 7.0 would result in the same percent contribution for the Tc-99 as shown above for the occupational inhalation pathway. The resin waste is the limiting material. The effluent concentration limit for effluent air is 9E-10 for Tc-99 or 330 times lower than the occupational DAC. In Section 8.3, the conservative occupational dose for resin waste is given as 0.4 mrem/yr, thus, the conservative effluent dose to the public from Tc-99 is less than 0.4/330 = 0.001 mrem/yr. Thus, this dose pathway to the public is negligible.

The Tc-99 dose from the water effluent can also be bounded. Section 4.0 of Ref.3.3 estimates the water effluent concentration is 466 pCi/L. Using the NRC effluent concentration limit for Tc-99 from Ref. 3.1 of 6E-5 microCi/ml (equivalent to 50 mrem/yr), the dose is 3.9 mrem/yr from the Tc-99. Thus, this dose pathway to the public is negligible. Any actual public exposure would be further reduced by the fact that the discharge is delivered directly to the river which would further dilute the activity concentration prior to public consumption.

There is no credible exposure pathway for oral ingestion of the waste materials by the public.

11.0 Computer Software

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

ENERCON Excellence—Every project. Every day.	CALCULATION PREPARATION CHECKLIST	CALC NO. EPM027-CALC-001
		REV. 0

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?	☒	☐	☐
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?	☐	☐	☒

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?	☐	☒	☐
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
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-Office 16 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.	☐	☐	☒
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?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
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:

A. Joseph Nardi 12/16/2019

Print Name and Sign Date