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Form 3.2-1

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MEYM		TIP 3.2 (Revision 2	<u>}</u>	Page: 1 of	24	
DCR NO (if applicable):		PROJECT NAME: High Capacity TN40HT Storage System			m	
PROJECT NO: T	N40HT	CLIENT: NMC			~~~	
CALCULATION TITLE: Dose Rates Estimate for Prairie Island ISFSI Comprised with TN40HT Casks Loaded with WE 14x14 OFA and WE 14x14 STD Fuel Assemblies.						
SUMMARY DESCRIPTION:						
1) Calculation Summary						
Calculate Prairie Island (PI) ISFSI d is assumed that 56 casks are loade WE 14x14 STD. Credit is taken loading. No control components considered.	d with WE 1 for radiolo	4x14 OFA Fuel Assemi gical source terms va	blies (FA) type riation in the	, the rest of the casks due to	e casks o schedu	contain iled ISFSI
2) Storage Media Description						1 CD
If original issue, is licensing rev	view per T	IP 3.6 required?				
Yes ☐ No ☒ (ex	plain belov	v) Licensing R	Review No.: _			
This calculation is to provide suppreview is not required.	ort for off-	site dose evaluation p	performed by	the client, NI	VIC. 100	FR72.48
Software Utilized:					Version:	
MCNP 5 v. 1.40					C0073	0MNYCP00
Calculation is complete:	a a					
Originator Name and Signature:	Terge	wid Terebure	evgenly Tereki	nlr	Date:	06/25/07
Calculation has been checked f	or consist	tency, completeness	s and correc	tness:		
Checker Name and Signature: Rick Migliore Date: 06/25/07						06/25 /07
Calculation is approved for use	•	Prak	act Nar	Manan		
Project Engineer Name and Signature	Makash Narayanan					S 16/07



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REVISION SUMMARY

REV.	DATE	DESCRIPTION	AFFECTED PAGES	AFFECTED DISKS
0	08/16/2007	Initial Issue	1 - 24	1



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1 Purpose

The purpose of this calculation package is to estimate the far field neutron and gamma dose rates in the vicinity of a customer, Prairie Island (PI), designed ISFSI. When fully loaded, the ISFSI admits 65 TN40HT Dry Storage casks arranged in two arrays. Each cask contains 40 fuel assemblies. The shielding calculation provides the dose rates as a function of distance (from the north-west corner of the array) at 600, 700 and 800 meters. Dose rates account for radiological source terms decrease due to scheduled loading at a pace of two casks per year until the ISFSI contains in total 65 casks.

2 References

- 1. TN Document, "Prairie Island ISFSI Safety Analysis Report", Revision 9.
- 2. TN Calculation "TN-40HT Near Field Shielding Calculation." TN40HT-0501 Revision 0.
- 3. NMC Letter, "Request for Proposal for More realistic PI ISFSI Offsite Dose Calculation", Letter Ref.# PINGP: 04FS03-A210-059
- 4. TN Calculation "TN-40HT Far Field Shielding Calculations." TN40HT-0502 Revision 0.
- 5. TN Calculation "Representative Source terms for the Prairie Island ISFSI." TN40HT-0510, Revision 0.
- "MCNP/MCNPX Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, January 2006.
- 7. "MCNP5 v1.40 Verification Test Report on the 64-Bit Opteron Running Linux." TN File No. E-23526, Revision 0,
- 8. Title 10, "Energy," Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation."
- 9. Title 10, "Energy," Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."
- 10. Title 40, "Protection of Environment," Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."



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- 11. "American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants," ANSI/ANS-6.6.1-1977, American Nuclear Society, Illinois, 1977.
- 12.TN Calculation "Standardized NUHOMS® System HSM Surface Dose Rates for High Burnup Fuel." NUH-HBU.0501, Revision 0.
- 13. Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III
- 14. "CASK-81 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis." ORNL DLC-23 Updated (June 16, 1981)
- 15. R.W. Roussin and J.B. Wright, "Contents, Energy Group Structure, and Weighting Function Used for DLC-23/Cask". Informal Notes (1972). DLC-0023/03
- 16.TN Calculation, "TN-40 Transport Fuel Qualification and Source Terms." 10421-012 Rev.1



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Assumptions 3

The shielding calculation is performed based on the following assumptions regarding

ISFSI layout, casks arrangement and MCNP models. The assumptions regarding under various sub-sections. Other assumptions are documented, wherever used in the calculation. The assumptions are based on data from reference [1] (Figure 1.3-1, Figure 2.5-6 and Figure 2.5-7 in particular from [1]) and [3].				
3.1 ISFSI Layout.				
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4 Methodology

The three-dimensional Monte Carlo particle transport computer code, MCNP, reference [6] has been utilized to calculate the dose rates. The MCNP computer code has been utilized by Transnuclear for shielding evaluations in NRC approved applications as shown in reference [1].

4.1 Source Determination

Source information required by MCNP includes neutron and gamma spectra, total neutron and gamma activities for each fuel assembly axial exposure zone and total neutron and gamma activities for the entire ISFSI. The neutron and gamma radiation source terms strength per fuel assembly and spectra are determined in reference [5].





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5 MCNP Analysis - Input and Outputs

The 3-D Monte Carlo code MCNP [6] is utilized to calculate the dose rates at the points of interest specified in item 4, Section 3.3.

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5.1 Geometry

Assumptions specified in Section 3 provide a "skeleton" for building ISFSI geometry in MCNP models. Casks in arrays are represented as lattice cells in MCNP input decks. Indices of the lattice cells can be mapped onto casks depicted on Figure 1 using Figure 2.



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5.2 Flux-to-Dose Rate Factors

Flux-to-dose rate factors for both neutrons and gammas are taken from reference [11]. MCNP logarithmically interpolates the flux-to-dose factor for each scoring particle from the tabulated values input in the DE and DF variables.

5.3 MCNP Input Files

Three MCNP runs are used to calculate the dose rates around the ISFSI.

- **GD.mi** and **NGD.mi** determines gamma radiation dose rate
- **ND.mi** and **ND 4.mi** determine the neutron radiation dose rate

The MCNP input and all associated output files(namely, *.mo, *_tl, *_tpe,) are listed in Section 8. They are also included on the attached compact disk. Note that information provided in the comment cards, either after \$ or "c" character (when "c" is the first character in a line) is optional and may not correspond to the actual parameter(s) used.

5.4 Calculations





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Form 3.2-1 Calculation Cover Sheet TIP 3.2 (Revision 2)

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DCR NO (if applicable) : NA	PROJECT NAME: High Capacity TN-40HT Storage Sys	tem
PROJECT NO: TN40HT	CLIENT: NMC	
CALCULATION TITLE: Representative Source Terms	s for the Prairle Island ISFSI	WWW. When the statement in the statement was a
SUMMARY DESCRIPTION:		
1) Calculation Summary		
The representative source terms as a function of the a site-specific ISFSI calculation.	ime are computed for both WE 14x14 STD and OFA fuel	to support
2) Storage Media Description		
1 CD		
If original issue, is licensing review per TIP 3.5	Positiva 2	personal state of the first fi
Yes ☐ No ☒ (explain below)		
	Licensing Review No.: to support the State of Minnesota Certificate of Need	Dranna to
Increase the storage at the Prairie Island ISFSI to	a total of 65 casks. This activity is not governed by the	FTQCess to ∋ NRC.
•		
Software Utilized:		Version:
SCALE		4.4
and the state of t		-T("V
Calculation is complete:	The second secon	
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X-3/1	·W -() -	4113/07
Orlginator Name and Signature: RJ Migliore	" lighter	Date:
Calculation has been checked for consistency,	completeness and correctness:	
	11 011	41307
Checker Name and Signature: SR Gardner	Sh R. G.	Date:
Calculation is approved for use:		
Prakash A Narayanan	A-Prakash	08/15/2007
Project Engineer Name and Signature:		Date:

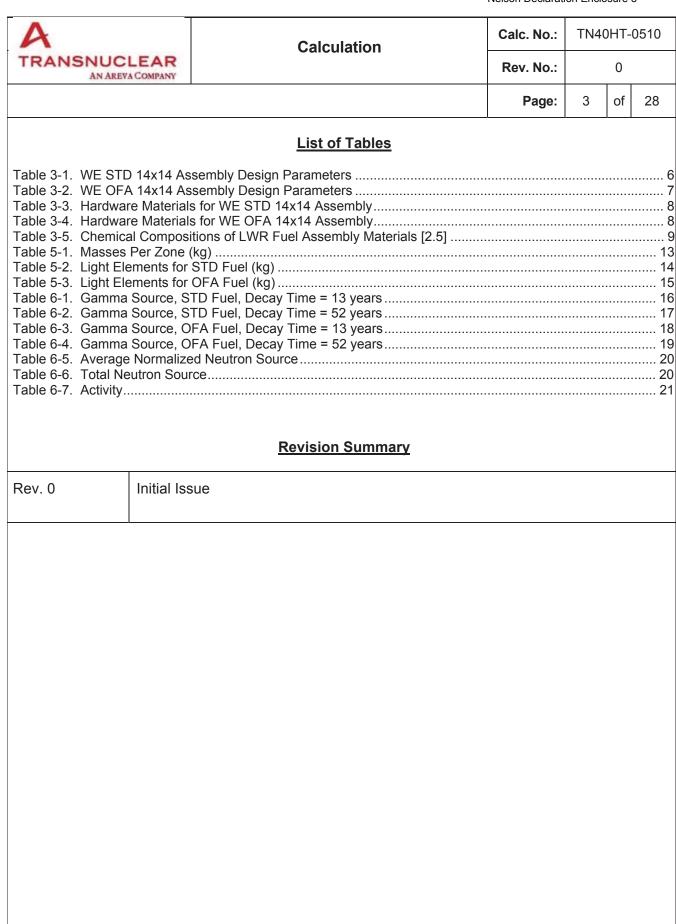


Calculation

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1.0 Purpose

The purpose of this calculation is to determine representative source terms for use in a site-dose calculation at Prairie Island. Both WE STD 14x14 (STD) and WE OFA 14x14 (OFA) fuel is considered.

2.0 References

- 2.1 SCALE-4.4, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," CCC-545, ORNL.
- 2.2 SCALE 4.4 Verification Test Report, Windows XP, TR-VV-07-003, Rev. 0. Packaging Technology, Inc.
- 2.3 DOE Report DOE/RW-0184-R1, Vol. 1, "Characteristics of Potential Repository Wastes," July 1992.
- 2.4 Luksic, PNL-6906, Volume 1, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal," June 1989.
- 2.5 ORNL/TM-11018, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," Oak Ridge National Laboratory, December 1989.
- 2.6 DOE/ET/47912-3 Vol. III, "Domestic Light Water Reactor Fuel Design Evolution," Prepared for U. S. Department of Energy Savannah River Operations Office, September 1981.
- 2.7 E-11402, Rev 3, "Design Criteria for the TN-40 Spent Fuel Storage Cask."
- 2.8 Transnuclear Calculation 1042-8, Rev. 1, "TN-40 Primary Gamma Shielding."
- 2.9 Transnuclear Calculation 1042-7, Rev. 0, "Source Terms for TN-40 Cask."
- 2.10 Transnuclear Calculation 10421-012, Rev. 1, "TN-40 Transport Fuel Qualification and Source Terms."
- 2.11 DCS E-22497, "Design Criteria for the TN-40HT High Burnup Spent Fuel Storage/Transportation Cask," Rev. 0.
- 2.12 NMC Letter From S. Leblang to M. Mason, "Weight of SS in Top End fitting of West. STD Fuel," Ref: PINGP: 04FS02-A210-015, Dated 8/2/05.
- 2.13 "SCALE-4 Analysis of Pressurized Water Reactor Critical Configurations: Volume 5-North Anna Unit 1 Cycle 5," ORNL/TM-12294/V5, 1996.

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3.0 Methodology

3.1 Design Inputs

3.1.1 Design Criteria

Important design parameters for this analysis are also provided in the assumptions Section 4 below. The TN-40HT transportation cask design criteria are set forth in the Design Criteria Specification [2.11].

3.1.2 Fuel Data

Fuel data for both the STD and OFA fuel are provided in Table 3-1 and Table 3-2, respectively. References for all values are provided in the tables.

Hardware materials and masses for both the STD and OFA fuel are provided in Table 3-3 and Table 3-4, respectively. The hardware masses for the STD fuel are consistent with those utilized in [2.10], which were primarily obtained from DOE/RW-0184-R1 [2.3].

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The masses for the OFA fuel are obtained from TN calculation 1042-7 [2.9] and were originally obtained from the client. The OFA fuel masses from [2.9] are not consistent with the masses in [2.3], although the masses utilized are considered representative of the actual fuel.

Fuel loading and burnup information to be utilized in this calculation was provided by letter from NMC to TN (this letter is reproduced in Appendix C).

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Isotopic compositions of common LWR fuel assembly hardware materials are taken from Reference [2.5] and listed in Table 3-5.

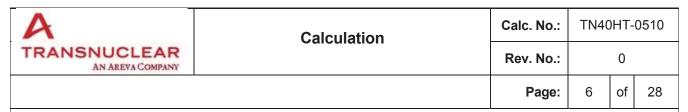


Table 3-1. WE STD 14x14 Assembly Design Parameters

Parameter	Value	Reference
Number of Rods	179	2.6
Overall Assembly Length (in)	161.1	2.6
Fuel Rod Length (in)	151.83	2.6
Plenum Length (in)	7.142	2.6
Active Fuel Zone (in)	144	2.6

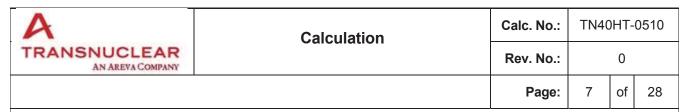


Table 3-2. WE OFA 14x14 Assembly Design Parameters

Parameter	Value	Reference				
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Fuel Rod Length (in)	151.85	2.3				
Plenum Length (in)	7.158	2.3				

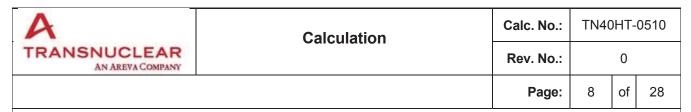


Table 3-3. Hardware Materials for WE STD 14x14 Assembly

Part Name	Parts per Assembly	Mass (kg)	Zone Material		Reference
Bottom Nozzle	1	7.893	Bottom	SS 304	2.3
Guide Tubes ^[3]	16	7.642	Core/Plenum	SS 304	2.3
Instrument Tube ^[3]	1	0.478	Core/Plenum	SS 304	2.3
Spacer-Incore	6	5.370	Core	Inconel-718	2.3
Spacer-Plenum ^[1]	1	0.680	Plenum	Inconel-718	2.3
Plenum Spring	179	5.684	Plenum	SS 302	2.3
Top Nozzle ^[2]	1	9.380	Тор	SS 304	2.3
Hold Down Springs	8	0.508	Тор	Inconel-718	2.3

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Table 3-5. Chemical Compositions of LWR Fuel Assembly Materials [2.5]

	Atomio	Material Composition, grams per kg of material						
Element	Atomic Number	Zircaloy-4	Inconel- 718	Inconel X- 750	Stainless Steel 304	UO ₂ Fuel (per kg U)		
Н	1	1.30E-02	-	-	-	-		
Li	3	-	-	-	-	1.00E-03		
В	5	3.30E-04	-	-	-	1.00E-03		
С	6	1.20E-01	4.00E-01	3.99E-01	8.00E-01	8.94E-02		
N	7	8.00E-02	1.30E+00	1.30E+00	1.30E+00	2.50E-02		
0	8	9.50E-01	-	-	-	1.34E+02		
F	9	-	-	-	-	1.07E-02		
Na	11	-	-	-	-	1.50E-02		
Mg	12	-	-	-	-	2.00E-03		
Al	13	2.40E-02	5.99E+00	7.98E+00	-	1.67E-02		
Si	14	-	2.00E+00	2.99E+00	1.00E+01	1.21E-02		
Р	15	-	-	-	4.50E-01	3.50E-02		
S	16	3.50E-02	7.00E-02	7.00E-02	3.00E-01	-		
CI	17	-	-	-	-	5.30E-03		
Ca	20	-	-	-	-	2.00E-03		
Ti	22	2.00E-02	7.99E+00	2.49E+01	-	1.00E-03		
V	23	2.00E-02	-	-	_	3.00E-03		
Cr	24	1.25E+00	1.90E+02	1.50E+02	1.90E+02	4.00E-03		
Mn	25	2.00E-02	2.00E+00	6.98E+00	2.00E+01	1.70E-03		
Fe	26	2.25E+00	1.80E+02	6.78E+01	6.88E+02	1.80E-02		
Co	27	1.00E-02	4.69E+00	6.49E+00	8.00E-01	1.00E-03		
Ni	28	2.00E-02	5.20E+02	7.22E+02	8.92E+01	2.40E-02		
Cu	29	2.00E-02	9.99E-01	4.99E-01	-	1.00E-03		
Zn	30	-	-	-	-	4.03E-02		
Zr	40	9.79E+02	_	-	_	-		
Nb	41	-	5.55E+01	8.98E+00	-	-		
Мо	42	-	3.00E+01	_	_	1.00E-02		
Ag	47	-	-	-	-	1.00E-04		
Cd	48	2.50E-04	-	_	_	2.50E-02		
In	49	-	-	-	_	2.00E-03		
Sn	50	1.60E+01	-	-	_	4.00E-03		
Gd	64	-	-	-	_	2.50E-03		
Hf	72	7.80E-02	_	_	_	-		
W	74	2.00E-02	_	-	_	2.00E-03		
Pb	82	-	_	_	_	1.00E-03		
U	92	2.00E-04	-	-	-	1.00E+03		

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These analyses were carried out using the SAS2H depletion module from the SCALE4.4 computer software [2.1] . All cases were run on PacTec machine ADANC491TC (Windows XP). SCALE4.4 is authorized for use on Windows XP machines [2.2]. For more detailed information on SAS2H and SCALE refer to the user manual. For all SAS2H calculations the latest SCALE 44 group ENDF/B-V (44groupndf5) library was used.

Several formulas were used to develop the SAS2H inputs. First, the weight fractions of the uranium isotopes were calculated using Equations 1 through 3 below [2.13]. Second, the active fuel length for the UO_2 was calculated using Equation 4 assuming 96% dense UO_2 . A proper active fuel length ensures the correct MTU is input to SAS2H. Third, the irradiation time was calculated using Equation 5.

 $W_{234} = 0.007731 (W_{235})^{1.0837}$ Equation 1

 $W_{236} = 0.0046 * W_{235}$ Equation 2

 $W_{238} = (1 - W_{235} - W_{236} - W_{238})$ Equation 3

FUELENGTH = M_U * 1000/[$\pi/4$ * OD² * 0.96 * 10.96 * 0.8814 * 179] Equation 4

where, M_U is the initial uranium loading in kilograms, 0.8814 is the weight fraction of U in UO_2 , 10.96 g/cm³ is the theoretical density of UO_2 , OD is the outer diameter of the pellet in centimeters, and 179 is the number of fuel rods in one assembly. For the OFA fuel, FUELENGTH = 360.80 cm, while for STD fuel, FUELENGTH = 355.16 cm.

$$T = BU * M_U / P$$
 Equation 5

where, T is the lifetime irradiation time in days, M_U is the initial uranium loading in metric tons, BU is the assembly burnup in MWD/MTU, and P is the power in MW/assy.

4.0 Assumptions

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4.3 Chemical impurities as defined in ORNL/TM-11018 [2.5] are assumed for the hardware and fuel materials. Flux scaling factors from PNL-6906 [2.4] are assumed for the top nozzle, bottom nozzle and plenum regions. These factors are: 0.2 – bottom nozzle, 1.0 – core, 0.2 – plenum, and 0.1 – top end fitting.

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4.4 The SS302 plenum springs are assumed to have a composition similar to SS304.								

5.0 Computation

Prior to development of the SAS2H models, the mass of light elements in each fuel zone must be determined. The fuel cladding, guide tubes, and instrument tube span both the core and plenum regions, so the mass of these items must be split appropriately.

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For the OFA fuel, this split has been performed in the reference document [2.9] (see Table 3-4) and no additional computations are required. Note that slightly different masses would be determined if these values were computed explicitly rather than taken from [2.9]. These differences are small and may be neglected. Details of the masses utilized are provided in spreadsheet *Materials – OFA.XLS*

Given the masses of each constituent listed in Table 5-1, the light element masses (including impurities) may be computed using the compositions listed in Table 3-5 and the scaling factors for each zone noted in Assumption 4.3. These light elements also include impurities in the UO_2 . The final light elements to be input to SAS2H are provided in Table 5-2 and Table 5-3 for STD and OFA fuel, respectively.

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Separate input files are developed for each of the four fuel zones. A listing of the input and output files is included in Appendix A. Input files are named with the following convention:

Type_zone#.in

where:

Type = OFA or STD

Zone = BN for bottom nozzle

CORE for in-core
P for plenum
TN for top nozzle

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Sample input files are included in Appendix B. The only differences between input files of the same fuel type are the light element composition and the output switch on the 84\$\$ card. (For the top, bottom, and plenum models, the 84\$\$ card is set so that only light elements are output.)

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A	Calculation	Calc. No.:	TN4)-THC	0510
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A	Calculation	Calc. No.:	TN40	THC	0510
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