ATTACHMENT 6

POST-ACCIDENT LETDOWN RADIATION MONITOR RESPONSE

- North Anna Calculation PA-0234, Rev. 1, Post-Accident Letdown Radiation Monitor Response
- Surry Calculation PA-0236, Rev. 0., Add. A, Post-Accident Letdown Radiation Monitor Response

Dominion Energy Nuclear Connecticut, Inc. (DENC)

Virginia Electric and Power Company
(Dominion Energy Virginia)

North Anna Calculation PA-0234, Rev. 1, Post-Accident Letdown Radiation Monitor Response

The following pertinent information has been extracted from North Anna Calculation PA-0234, Rev. 1, Post-Accident Letdown Radiation Monitor Response. It is provided to assist technical reviewers that will be evaluating this license amendment request.

Purpose:
This calculation provides the setpoint as a function of decay of the letdown line radiation monitor during accident condition for a 1% and 5% failed fuel (gap release) and iodine spike cases of 60 µCi/gm and 300 µCi/gm Dose Equivalent I-131 released coolant activity in the RCS. During accident conditions, letdown radiation monitors support in determining the fuel failures that correspond to specific radiological criteria in the Emergency Action Levels (EALs).

References:

1. RF- DCP-000, 59-DCP-07-006, "Letdown Radiation Monitor Replacement / North Anna / Unit 1 & 2."
2. *Nuclear Energy Institute NEI 99-01, Rev. 4, "Methodology for Development of Emergency Action Levels," January 2003.
3. PA-DWG-000, 11715-FM-095A, Rev. 28, "Flow/Valve Operating Numbers Diagram Chemical and Volume Control System," North Anna Power Station - Unit 1.
4. RF-DCP-000, 59-DCP-94-013, "Letdown Radiation Monitor Replacement - North Anna Unit 1," September 1995.
5. RF- DCP-000, 59-DCP-94-014, "Letdown Radiation Monitor Replacement - North Anna Unit 2," March 1995.
6. RF - CALC-NFL, PA-0195, Rev. 0, "Radiological Consequences of Fuel Handling Accident at North Anna Based on the Alternative Source Term," June 2003.
7. RF - CALC-NFL, PA-0219, Rev. 0, "Post Accident Response Curves for North Anna Primary Hot Leg Sample Lines," January 2005
8. RF - CALC-MEC, ME-0438, Rev. 2 "Reactor Coolant Letdown Radiation Monitor Setpoints," June 1995.
9. *PA-REF=000, NUREG-1228, "Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents," McKenna, T. J. and Gutter, U .S. Nuclear Regulatory Commission, Washington, D .C, 1988
10. *PA-REF-0, REG. Gu-1 .109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR PART 50, Appendix I," Rev. .1, U.S. Nuclear Regulatory Commission, Washington, D .C, 1977.
11. RF-CODE-000, MicroShield, Grove Software, Inc, Verification 7 .02.
12. RF - CALC-RAD, PA-0246, Rev. 1, "Letdown Radiation Monitor Setpoint for North Anna," Apri1 2008.

Computer Codes Used:
MicroShield Version 7.02 (Ref. 11)

Method of Analysis:
The 1% failed fuel is modeled in this calculation by calculating the dose rates resulting from 1% of the failed fuel gap inventory being released into the primary coolant that resulted following an accident. The 5% failed fuel was modeled by scaling up by a factor of 5 the results for the 1% failed fuel.

Dose rates to the monitor are calculated using MicroShield computer code. (Detector assembly is placed approximately 2 .5 inch (including 0 .5 inch of insulation of the pipe) from the center of a 2-inch diameter pipe source. The use of the MicroShield code is reasonable since the source is surrounded by lead shield and scattering is considered to be minimal. The MicroShield modeling assumes liquid source in a 1-inch radius pipe and 10 inches long with a thickness of 0 .154 inches. The dose point is assumed to be located at 2 .5 inch from the center (radial direction) of the source. The pipe and the detector are placed inside lead shield wall. It is assumed that the detector is located in the middle of the length of the pipe.

Conclusion:
Tables 4 below provides a summary of results of the letdown radiation monitor dose rates that can be used for the EAL radiological criteria.

The dose rates are calculated for the letdown line.

Table 4
North Anna Letdown Radiation Monitor Dose

Time (hr.)	Dose (R/hr.)	Dose (R/hr.)	Dose (R/hr.)	Dose (R/hr.)
	1% FF	5% FF	60 μCi /gm	300 μCi /gm
0.0	46.85	234.25	24.85	124.26
1.0	28.65	143.25	15.20	75.99
2.0	20.94	104.70	11.11	55.54
4.0	14.20	71.00	7.53	37.66
8.0	9.30	46.50	4.93	24.67
16.0	6.03	30.15	3.20	15.99
24.0	4.76	23.80	2.52	12.62
36.0	3.86	19.30	2.05	10.24
48.0	3.41	17.05	1.81	9.04
72.0	2.97	14.85	1.58	7.88

Calculation PA-0236, Rev. 0., Add. A, Post-Accident Letdown Radiation Monitor Response for Surry

The following pertinent information has been extracted from Calculation PA-0236, Rev. 0., Add. A, Post-Accident Letdown Radiation Monitor Response for Surry. It is provided to assist technical reviewers that will be evaluating this license amendment request.

Purpose:
The purpose of this addendum is to determine the letdown radiation monitor (RM) response to source terms representative of gross reactor coolant activity and fuel failure that correspond to specific radiological criteria in the Emergency Action Levels (EALs) based on guidance from the Nuclear Energy Institute (NEI) for primary coolant activity level.

References:
1. NEI 99-01, Rev. 4, "Methodology for Development of Emergency Action Levels," January 2003.
2. NEI 99-01, Rev. 6, "Development of Emergency Action Levels for Non-Passive Reactors," November 2012.
3. Computer Code MicroShield, "MicroShield User's Manual," Grove Software, Inc.
4. MICROSHIELD-20170323-0-0, "MicroShield V. 7.02 Periodic Effectiveness Review 2017, Code Manager Change and Code Owner Change."
5. 958.398ABS Rev. A, "Calibration of a 903664 Letdown Monitor and 943-36 Detector," Victoreen, Inc., June 1996.
6. 11448/11548-7.57-14B Sheet 1, "Hi & Lo Range Letdown Monitor," June 1970 (Detector Shield Arrangement – Vendor Drawing 903664).
7. 11448/11548-7.57-26A Sheet 1, "Connections of Letdown Monitor," June 1970 (Sample line tubing details – Vendor Drawing 903742).
8. 11448/11548-7.57-26B Sheet 1, "Connections of Letdown Monitor," June 1970 (Sample line tubing details – Vendor Drawing 903742).
9. 903787, "[Aluminum] Spacer Hi-Lo," August 1970.
10. 903751 Rev. X1, "[Lead] Plug," August 1970.
11. RA-0008 Rev. 0 through Addendum C, "Core Isotropic Inventories for Surry Dose Consequences Analyses Based on the Alternate Source Term," July 2010.
12. ETE-NAF-2017-0052 Rev. 1, "Input to a Proposed Surry License Amendment Request Adopting TSTF-490-A Rev. 0 (Dose Equivalent Xe-133) and Updated Alternative Source Term Analyses," January 2018.
13. NUREG-1228, "Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents," McKenna, T. J. and Gutter, U.S. Nuclear Regulatory Commission, Washington, D.C, October 1988.

14. RA-0070 Rev. 0 through Addendum A, "Radiological Consequences of a Steam Generator Tube Rupture (SGTR) at Surry Power Station Based on the Alternative Source Term (AST)," May 2017.
15. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors For Inhalation, Submersion, and Ingestion," EPA 520/1-88-020, September 1988.
16. SQA-WATPROP-D-20170608, "SQA Documents for the Initial Release of WATPROP-D and Initial Periodic Effectiveness Review," June 2017.

Computer Codes Used:
MicroShield Version 7.02 [References 3 and 4]

Methodology:
Letdown radiation monitor response in counts per minute (cpm) was determined for accident source terms of 1% failed fuel, 10 μCi/cc DE I-131, and 300 μCi/cc DE I-131 at 0, 1, 2, 4, 8, 16, 24, 36, 48, and 72 hours after shutdown. This was accomplished by using MicroShield to determine dose rate at the detector for the 1% failed source term at 0 hours. MicroShield was used to decay the 1% failed fuel source and determine the dose rates at the detector for each of the subsequent time steps. MicroShield dose rates were also determined for calibration standards of Co-60 and Mn-54 for which the vendor had determined cpm. Conversion factors for cpm to Dose rate for Co-60 and Mn-54 were derived and applied to the dose rate results for the 1% failed fuel source term at each time step.

Conclusion:
The letdown radiation monitor response in counts per minute (cpm) to accident source terms of 1% failed fuel, 10 μCi/gm DE I-131, and 300 μCi/gm DE I-131 at 0, 0.5, 1, 2, 4, 8, 16, 24, 36, 48, and 72 hours after shutdown are documented in Table 4.

Table 4: Surry Letdown Radiation Monitor Response (cpm)

Time (hr.)	1% FF	10 μCi/gm	300 μCi/gm
0.0	1.30E+07	1.18E+06	3.53E+07
0.5	5.98E+06	5.40E+05	1.62E+07
1.0	4.02E+06	3.62E+05	1.09E+07
2.0	2.33E+06	2.11E+05	6.32E+06
4.0	1.32E+06	1.19E+05	3.57E+06
8.0	7.05E+05	6.37E+04	1.91E+06
16.0	3.08E+05	2.78E+04	8.35E+05
24.0	1.81E+05	1.64E+04	4.91E+05
36.0	1.16E+05	1.04E+04	3.13E+05
48.0	9.34E+04	8.43E+03	2.53E+05
72.0	7.94E+04	7.17E+03	2.15E+05