FENOC™
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077

Eric A. Larson
Site Vice President

724-682-5234
Fax: 724-643-8069

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L-14-258

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Response to Request for Supplemental Information Regarding Spent Fuel Storage Pool Minimum Inadvertent Drainage Elevation License Amendment Request (TAC No. MF4213)

By correspondence dated June 2, 2014 (Accession No. ML14153A388), FirstEnergy Nuclear Operating Company (FENOC) submitted to the Nuclear Regulatory Commission (NRC) a proposed amendment to the Beaver Valley Power Station, Unit No. 2, Technical Specifications (TSs). The proposed amendment is required to correct the minimum drain elevation for the spent fuel storage pool specified in Technical Specification 4.3.2.

By correspondence dated July 24, 2014 (Accession No. ML14190A684), the NRC staff requested supplemental information to support FENOC's June 2, 2014 correspondence. Attachment 1 provides a response to the NRC's July 24, 2014 request.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 8, 2014.

Sincerely,

Eric A. Larson

Attachment:
Response to July 24, 2014 Request for Supplemental Information

cc: NRC Region I Administrator
NRC Senior Resident Inspector
NRR Project Manager
Director BRP/DEP
Site BRP/DEP Representative

By letter dated June 2, 2014 (Accession No. ML14153A388), FirstEnergy Nuclear Operating Company (FENOC) requested an amendment to the operating license for Beaver Valley Power Station, Unit No. 2 (BVPS-2). The proposed amendment would correct the minimum drain elevation for the spent fuel storage pool specified in BVPS-2 Technical Specification 4.3.2, "Drainage."

In a July 24, 2014 letter (Accession No. ML14190A684), the NRC staff requested that FENOC supplement the application dated June 2, 2014. The information requested is provided below in bold text, followed by the FENOC response.

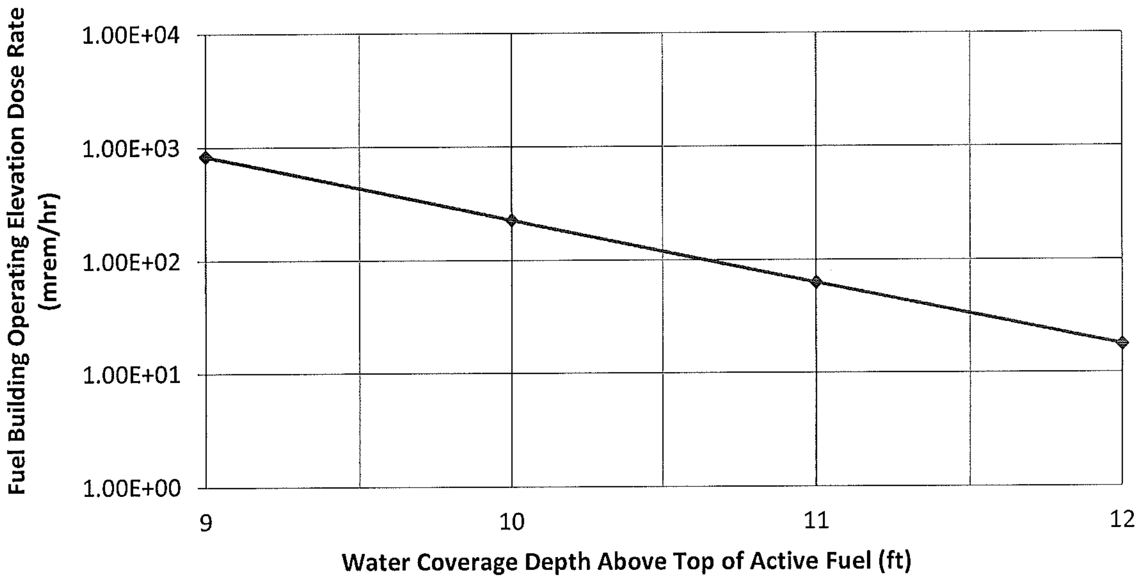
1) A calculation sheet which cites the appropriate Regulatory Guide methodology and provides the calculation that demonstrates an estimated dose rate of approximately 280 millirem per hour for a person located at the fuel building operating floor elevation, directly above freshly discharged fuel assemblies stored in the spent fuel storage pool with the pool water level reduced to the 750 feet, 10 inch elevation.

Response:

Summarized below are the results, methodology, critical design inputs, and assumptions presented in the analysis of record that determined the dose rates at the fuel building operating floor elevation for reduced water coverage above the active fuel in the spent fuel storage pool racks. This summary is provided rather than the requested calculation sheet as agreed to by the NRC staff during a telephone conference on July 23, 2014. The objective is to provide the information necessary to confirm the dose calculation results cited in the June 2, 2014 letter and described in the information request above.

Figure 1 presents the calculated dose rate at the fuel building operating floor elevation, directly above the freshly discharged fuel assemblies, as a function of water coverage above the top of the active fuel.

Figure 1
Dose Rate at Fuel Building Operating Floor Elevation, As a Function of Water Coverage



The exponential function least squares curve fit of the Figure 1 dose rate versus water coverage data is given in Equation 1.

$$DR(x) = (8.3326E07)\, e^{-1.2806x} \qquad \textit{Equation 1}$$

Where *DR* is the dose rate (in mrem per hour) at the fuel building operating floor elevation, and x is the water coverage depth above the top of the active fuel in feet.

The dose rate at the fuel building operating deck elevation, 767 feet – 10 inches, is estimated by Equation 1 to be approximately 280 millirem per hour with the active fuel water coverage decreased to 9.84 feet (3 meters). Three meters of water coverage was selected for evaluation because it bounds the minimum water level of approximately 9.89 feet over the top of the active fuel that is present when the spent fuel storage pool is drained to elevation 750 feet – 10 inches.

The methodology principally involved in calculating the fuel building dose rate involves the isotopic depletion and decay application of the SCALE code system (including ORIGEN), and shielding code SW-QADCGGP. Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," July 2000, Section 3.1, "Fission Product Inventory," states on page 1.183-12 that the core inventory should be determined using an appropriate isotope generation and depletion computer code, such as ORIGEN 2 and ORIGEN-ARP. The shielding code QAD is among the acceptable shielding codes listed in Subsection 2, "Shielding," page 12.3-8 of NUREG-0800, Sections 12.3 - 12.4, "Radiation Protection Design Features," Revision 2.

The critical design inputs and assumptions that resulted in the development of the Figure 1 dose rate curve are as follows:

(1) Loss of spent fuel storage pool water level was assumed to occur after a normal full core offload to the spent fuel storage pool. The radioactivity inventory in the off-loaded core was conservatively based on a decay time of T = 100 hours (that is, the earliest time allowed for fuel movement after shutdown per the BVPS-2 Licensing Requirements Manual, without crediting decay during fuel transfer).

(2) The radiation source in the spent fuel storage pool includes 157 freshly discharged fuel assemblies from a full core offload and 72 fuel assemblies from the previous refueling outage that have been decayed for 18 months. Engineering judgment indicates that the dose contribution due to other previously discharged fuel assemblies (that is, older fuel assemblies in the spent fuel storage pool with decay times greater than 18 months) will be insignificant compared to the contribution from the freshly discharged fuel assemblies plus the 72 fuel assemblies that have been decayed 18 months.

(3) The fuel assemblies were arranged to maximize the dose rates along the centerline of the storage area; specifically, the 157 freshly discharged fuel assemblies were placed contiguously in a square block with the 72 fuel assemblies that have been decayed 18 months at the periphery of the block.

(4) The gamma energy emission rate per assembly for the 100 hour decayed fuel assemblies and the 18 month decayed fuel assemblies were calculated using SCALE4.3-ORIGEN-S and the BVPS-2 core inventory at the end of an equilibrium fuel cycle. The BVPS-2 licensing basis equilibrium core inventory was calculated by SCALE4.3 control module SAS2H and was based on:

- Full power operation at the licensed core power level (2918 MWt, including power uncertainty margin)
- U-235 enrichments ranging from 4.2 percent to 5 percent
- Core inventory calculated for 18-month fuel cycles with 518-day operation and assuming a 30 day refueling shutdown
- Maximum burnup of 40.225 MW/MTU

(5) To account for the variation of power level among fuel assemblies when they were irradiated in the core and the fact that the centrally located fuel assemblies will contribute more dose rate than those located peripherally, a conservative radial peaking of 1.3 was assumed for all 157 freshly discharged fuel assemblies and the 72 fuel assemblies that have been decayed 18 months. This radial peaking factor is conservative because all 157 freshly discharged fuel assemblies are included in the model, and the dose point – source configuration is such that a large number of fuel assemblies contribute to the dose rate of interest. (Note: The radial peaking factor of the worst fuel assembly is 1.5.)

(6) The gamma radiation source was homogenized in a storage cell. The minimum pitch between two storage cells is 8.977 inches. The volume that contains the radiation source in a fuel assembly is 8.977 inches by 8.977 inches by 144 inches. The homogenized gamma source ($MeV/sec/cm^3$) for the fuel assemblies decayed 100 hours and the fuel assemblies decayed 18 months are presented in Table 1.

(7) The axial distribution of the radiation source along the active fuel length was assumed to be uniform. This assumption is conservative in calculating the radiation dose above the fuel assemblies.

(8) The dose receptor location was conservatively assumed to be directly above the freshly discharged fuel assemblies, at the elevation of the operating floor, along the centerline of the freshly discharged fuel assembly region. The distance between the top of the active fuel stored in the racks and the elevation of the operating deck is approximately 27 feet. Gamma dose rates were calculated with water levels at 9, 10, 11, and 12 feet above the active fuel stored in the racks. The dose rates were calculated by the point-kernel ray-tracing shielding code SW-QADCGGP. Conservative build-up factors were used, and the model was prepared to ensure that un-accounted streaming/scattering paths were eliminated. American National Standard ANSI/ANS 6.1.1-1977 dose conversion factors were employed to convert the calculated gamma flux to dose rate.

Table 1, Gamma Source in the Homogenized Spent Fuel Storage Cell

Group midpoint energy (MeV)	100-hour decay homogenized source ($MeV/sec/cm^3$)	18-month decay homogenized source ($MeV/sec/cm^3$)
1.00E-02	1.43E+10	6.49E+08
2.50E-02	5.14E+09	3.36E+08
3.75E-02	1.13E+10	6.13E+08
5.75E-02	9.08E+09	6.81E+08
8.50E-02	2.12E+10	7.90E+08
1.25E-01	8.82E+10	1.49E+09
2.25E-01	1.00E+11	1.90E+09
3.75E-01	7.18E+10	1.57E+09
5.75E-01	3.04E+11	2.59E+10
8.50E-01	5.21E+11	1.43E+10
1.25E+00	6.31E+10	2.43E+09
1.75E+00	3.23E+11	4.10E+08
2.25E+00	2.04E+10	8.45E+08
2.75E+00	1.89E+10	1.71E+07
3.50E+00	2.03E+08	2.64E+06
5.00E+00	1.07E+03	7.76E+02
7.00E+00	1.72E+02	1.25E+02
9.50E+00	2.69E+01	1.95E+01
Sum	1.57E+12	5.19E+10

(9) Since minimizing water density is conservative in calculating the doses shielded by water, a water density of 0.95 g/cm^3 in the spent fuel storage pool was assumed for the evaluation.

(10) The SW-QADCGGP model consists of a homogenized source region that includes the active fuel region of fuel assemblies and the interstitial water medium; a fuel rod plenum region that includes the Zirlo cladding and the interstitial water; the water shield region above the fuel assembly top nozzle; and the air from the fuel pool surface to the operating floor elevation. The density of the homogenized source region is 3.929 g/cm^3, with major constituents of uranium, zirconium, oxygen and hydrogen, and a small amount of niobium, tin and iron. The density of the fuel rod plenum region is 1.17 g/cm^3, with major constituents of zirconium, oxygen and hydrogen, and a small amount of niobium, tin and iron. The height of the rod plenum region is 9.22 inches. The shielding provided by fuel assembly top nozzles and the storage racks was conservatively ignored. The water coverage depth (above active fuel) in Figure 1 includes the rod plenum region of 9.22 inches.