SNUPPS

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Power Plant System

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Nicholas A. Petrick
Executive Director

October 22, 1984

SLNRC 84-124 FILE: 0540
SUBJ: Neutron Shielding Water Can Activation

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket No: STN 50-482

Ref: SLNRC-0088, May 31, 1984

Dear Mr. Denton:

The referenced letter requested a partial exemption to GDC-4 to allow the water bags used for reactor cavity shielding to be replaced with thin-walled stainless steel water cans. The NRC, in its review of the referenced letter, asked questions about anticipated radiation doses to plant operating and maintenance personnel resulting from activation of the stainless steel can material. This letter describes our calculation of the activation of the stainless steel and the resultant dose rates. Calculated dose rates are higher than reported via telephone to Mr. Block because actual cobalt content has been determined to be higher than originally assumed. The assumptions used in the calculations are conservative and hence the actual dose rates should be lower. Upon approval by the NRC of the exemption request of the referenced letter, an FSAR change incorporating the results of this analysis will be processed.

Method of Analysis

1) The total neutron "flux" (5.77×10^7 neutrons/cm^2-sec) directly below the shield cans, resulting from a Monte Carlo calculation performed by Bechtel, was used as the starting point for the activation analysis. This flux was treated as a mono-directional epithermal neutron current entering the bottom face of the can, and was assumed to exist over the entire bottom face. This is conservative since not all neutrons that comprise the calculated flux will enter the can, and some of these neutrons will be fast rather than epithermal.

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2) All activation was assumed to be in the bottom face of the can. Since the top face is 12 inches from the bottom, and the entering neutrons undergo a factor of 20 attenuation in traversing the can, neutron activation in the top face of the can can be neglected.

3) All non-thermal neutrons that enter the can were assumed to be thermalized within the can, i.e., epithermal neutron leakage from the can during thermalization was assumed to be zero. This is conservative by 20 to 30 percent.

4) Net thermal neutron leakage from the can was assumed to be zero. This assumption more than compensates for the thermal neutron flux that enters the can from the adjacent concrete biological shielding, since the neutron source in the concrete is lower than in the shielding can.

5) Calculated dose rates were based on the saturated activity of radioactive isotopes at the time of reactor shutdown. This is valid for Mn-56, Cr-51, and Fe-59, but is conservative for Co-60 over most of the life of the plant.

Calculated Dose Rates

The following table summarizes the bottom surface dose rate for the four isotopes evaluated.

Active Isotope	Surface Dose Rate (mr/hr) 1 day	10 days	100 days
Co-60	56.6	56.6	56.6
Mn-56	0.6	0	0
Cr-51	3.4	2.7	0.3
Fe-59	2.6	2.3	0.6
Approximate Total Dose Rate	63.2	61.6	57.5

Very truly yours,

N A Petrick

Nicholas A. Petrick

FS/bds/13b7

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