TESTIMONY OF ROGER W. SINDERMAN CONCERNING RADIATION DOSES TO THE PUBLIC FROM FUEL STORED NEAR THE SOUTH WALL OF THE SPENT FUEL POOL CHRISTA-MARIA CONTENTION 2 AND O'NEILL CONTENTION IIA

I. Introduction

The purpose of this testimony is to respond to Christa-Maria Contention 2 and O'Neill Contention IIA by showing that spent nuclear fuel stored near the tapered south wall of the spent fuel pool will not result in radiation dose rates off-site in excess of regulatory limits. Consumers Power Company has performed calculations describing radiation doses to the nearest off-site location as a result of radioactive shine from spent fuel stored adjacent to the tapered south wall.[1/] Two cases were considered as follows:

1. The existing fuel channel rack remains in place as shown in Exhibit 1 and the location of fresh spent fuel is limited to that portion of the tapered wall occupied by the new type E Rack shown in the same Figure.

2. Same as above, but only fuel having decayed one year is placed in new rack E.

II. Discussion

Calculations for each case described were made taking into consideration the thickness of the tapered wall,

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1/ These calculations are set forth in Sinderman Exhibit 2.

the nearest distance to the site property boundary in line with the direction of the radiation exposure, the shielding afforded by the containment building, and the shielding of the air between the containment building and the site boundary. Radiation attenuation caused by forest growth as well as that produced by the auxiliary building structure was neglected to afford a degree of conservatism in the results obtained.

As has been previously indicated in Mr. Axtell's testimony concerning Christa-Maria Contention 2 and O'Neill Contention II-A, Consumers Power Company has chosen to keep the existing fuel channel rack (Rack C of Sinderman Exhibit 1) at its present location. This ensures that spent fuel cannot be stored near the thinnest section of the south wall of the spent fuel storage pool. Further, only fuel having decayed for at least one year after removal from the reactor will be stored in the outer three rows of the new Type E storage rack (those rows most closely adjacent to the fuel pool wall). Therefore, the results for Case Number 2 represent the actual conditions. Case 1 is, however, presented to show that radiation dose rates at the site boundary would not be significant even if new spent fuel were to be stored in the "E" rack next to the tapered portion of the south wall.

The dose rates immediately outside the south wall of the pool have been calculated by NUS, as described in the testimony of William Bell. From this one can calculate the radiation dose rate at the site boundary, as follows.

The spent fuel in the "E" rack was represented by a disk source with an area equal to the area formed by the vertical plane of the array of spent fuel in the "E" rack. The radius of this disk was calculated to be 3.45 feet. The disk source term was calculated as that necessary to produce the dose rates calculated by NUS for the outside of the south wall (that is, through an average 5.25 feet of concrete shielding). Accounting for attenuation by the containment structure and intervening air, the appropriate formula to be used for calculating radiation dose at the site boundary is:

$$R_{\gamma B} = \frac{R_{\gamma w}}{\frac{2 \ln(w^2+r^2)}{w^2}} \ln \frac{r^2+d^2}{d^2} B_a B_c e^{-(\mu_c t_c+\mu_a d)}$$

Where: $R_{\gamma B}$ is the dose rate in millirem per hour at the site boundary

w is the pool wall thickness in feet for each case

r is the disk radius in feet for each case

$R_{\gamma\omega}$ is the dose rate in millirem per hour at the outside of the pool wall in each case

d is the distance to the site boundary (2900 feet)

B_a and B_c are the gamma build-up factors for air and containment respectively

μ_a and μ_c are the mass absorption co-efficients for air and containment steel respectively

and t_c is the thickness of the containment building steel (3/4 of an inch)

See Hine and Brownell, RADIATION DOSIMETRY, Academic Press (1961), Chapter 16.

The results of the calculations for each case are shown below:

TABLE I

RADIATION DOSES AT THE SITE BOUNDARY RESULTING FROM SPENT FUEL STORED ADJACENT TO THE SOUTH FUEL POOL WALL

CASE	ANNUAL DOSE AT SITE BOUNDARY	REGULATORY LIMITS 10 CFR 20	REGULATORY LIMITS 10 CFR 50 APPENDIX I
1	0.022 mr/yr	500 mr/yr	10 mr/yr
2	1.6 E-04 mr/yr	500 mr/yr	10 mr/yr

The regulatory limit of 10 millirem/yr established by 10 CFR Part 50 Appendix I is a limit on plant <u>effluents</u>. It applies to doses due to releases of radioactive materials, and not to doses due to radioactive shine such as those calculated in this testimony. There are no releases of radioactive materials to the environment through the south wall of the Big Rock Point Plant spent fuel pool. Therefore the Appendix I limit referred to in Christa-Maria Contention 2 and O'Neill Contention IIA is not applicable, although it is shown in Table I for illustrative purposes only.

III. Conclusions

In both cases, the dose at the site boundary due to shine through the south wall of the spent fuel pool is within the regulatory limits of 10 CFR 20 and the design objectives of 10 CFR 50, Appendix I. In keeping with the constraints of maintaining radiation exposures to both the workers and the general public as low as reasonably achievable (ALARA), however, Consumers Power Company will insure that (only) spent fuel having decayed for at least one year will be stored in the outer three rows of new rack E.

TYPE OF RACK	NO. OF RACKS	NOMINAL CAPACITY	RACK STATUS	ACTUAL SPACES	TYPE OF STORAGE
* A	2	6 x 8	EXISTING	96	FUEL & INCORES
B	1	6 x 12	EXISTING	72	FUEL & CONT. BLADES
C	1	9 x 10	EXISTING	90	FUEL CHANNELS ONLY
D	1	8 x 11	NEW	88	SPENT FUEL
E	1	9 x 9	NEW	81	SPENT FUEL
F	1	8 x 13	NEW	104	SPENT FUEL

* SPENT FUEL WILL NOT BE STORED IN TYPE "A" RACKS DURING CASK HANDLING OPERATIONS

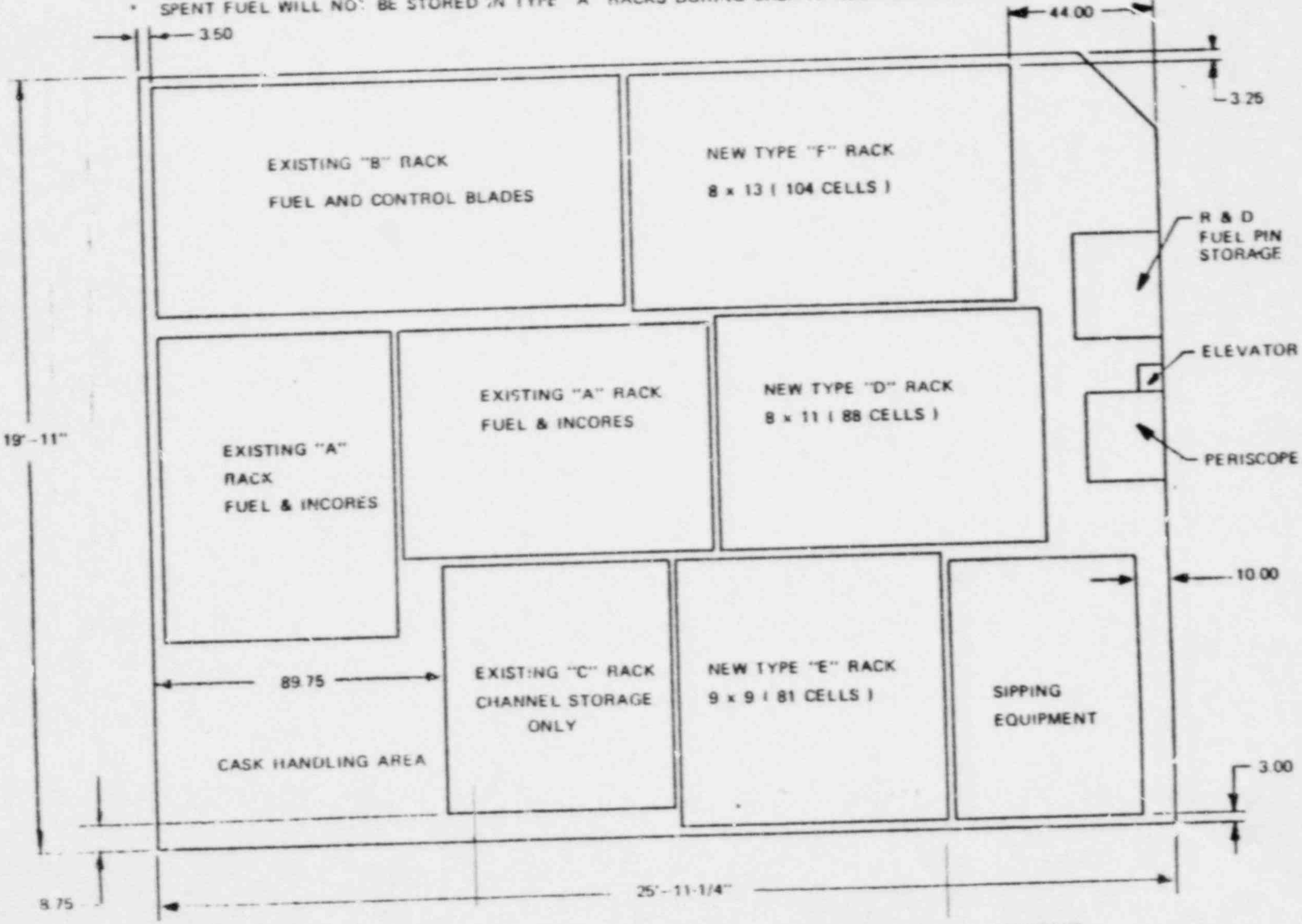


FIGURE 2–1 BIG ROCK POINT SPENT FUEL STORAGE RACK ARRANGEMENT

SINDERMAN EXHIBIT 1
(CHRISTA-MARIA CONTENTION 2 AND O'NEILL CONTENTION IIA)

To RWSinderman, P-24-115B

From SKBanbury, P-24-114

Date August 24, 1981

Subject DOSE RATE AT THE SOUTHWEST SECTOR SITE BOUNDARY FROM THE SPENT FUEL POOL-BIG ROCK POINT PLANT

Consumers Power Company

Internal Correspondence

CC RMMarusich, P-24-110

SSB-81-13

The following calculations were performed by the writer at your request. A technical review by RMMarusich note some conservatisms in the calculation; shielding provided by the control room wall and the containment wall thickness being greater than the 3/4" thickness used due to the angle of the rays.

A detailed description of the calculation methods used and resulting data is attached. Final results listed below:

Dose rate at the site boundary from fuel stored with one year decay is 1.58E-04 mr/yr.

Dose rate at the site boundary from newly irradiated stored fuel is 2.17E-02 mr/yr.

(CHRISTA-MARIA CONTENTION 2 and O'NEILL CONTENTION IIA)

Control Room Wall in Relation
to Spent Fuel Pool

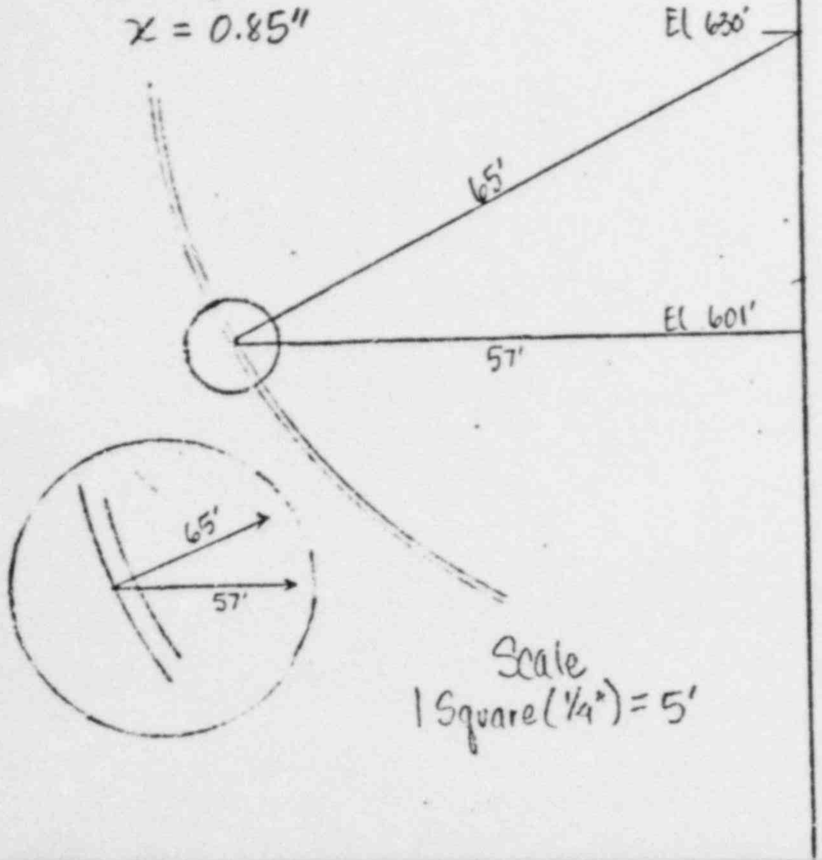
Scale
1 square (1/4") = 10'

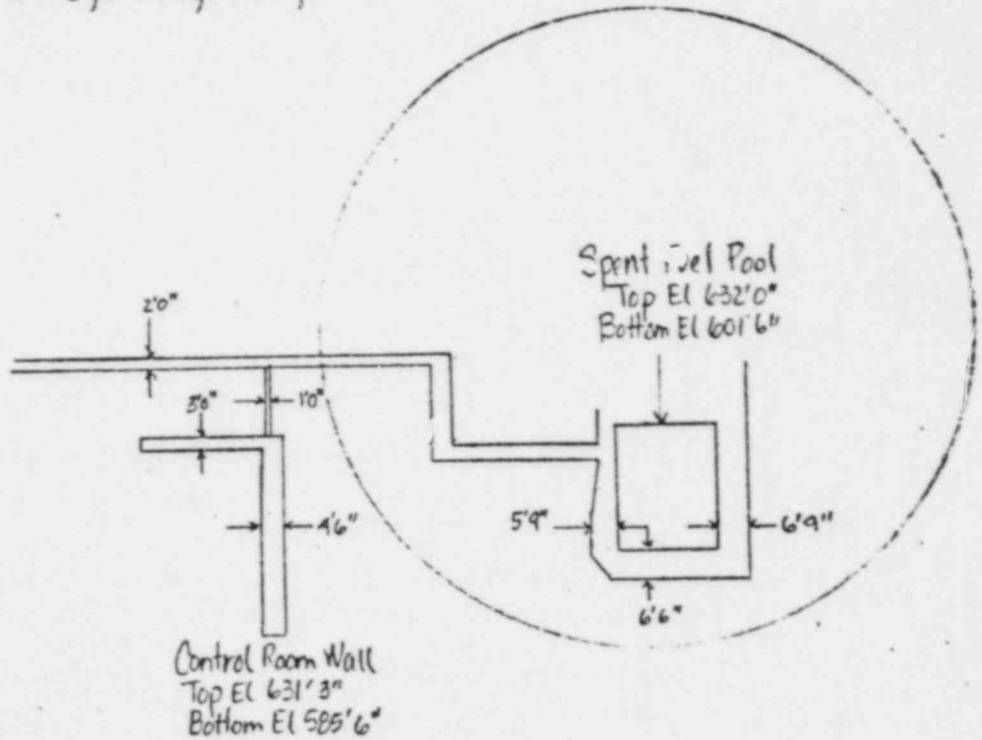
Plan above operating floor
Reference Dwgs C-113, C-253, M-101, M-103

Containment thickness
by similar triangles

$$\frac{x}{65} = \frac{3/4}{57}$$

$$x = 0.85''$$





Dose to person at site boundary from spent fuel area.

Reference RAE memo 25-79 (attached), the dose rate at El 600'6" is 38 mR/hr.

Scaled Dwg C-3 to determine distance from spent fuel area to site boundary in southeast sector.

Approximately 2900' SW Sector distance — 2680' nearest property boundary line per technical specifications

First assuming the source is a point source determine dose rate at site boundary without shielding.

From RAE calculation used 4.1' (greatest distance therefore most conservative) as the distance from source to dose point which obtained the dose rate of 38 mR/hr outside the spent fuel pool area.

$$\frac{4.1^2}{2900^2}(38 \text{ mR/hr})\frac{(24)(365)\text{hrs}}{\text{yr}} = \text{mR/yr @ boundary}$$

$$= 0.67 \text{ mR/yr @ boundary without shielding}$$

Second determine dose rate using the shielding provided by the containment shell and air.

For determining mass attenuation coefficients and buildup will assume 1 MeV ray.

Containment shell is 3/4" iron

$cm = 1.91$
$\rho = 7.86\ g/cm^3$
$\mu/_\rho = 0.0599\ cm^2/g$
$\mu\chi = 0.894$

Assume $\mu x = 1$ for buildup

$B = 1.87$

2900' of air
34,800 inches of air

$cm = 88{,}393$
$\rho = 0.001293\ g/cm^3$
$\mu/_\rho = 0.0636\ cm^2/g$
$\mu\chi = 7.26$

To find buildup of air will use average of buildups of aluminum and water

$B_{water} = 17.1$

$B_{aluminum} = 13.8$

$$\frac{17.1 + 13.8}{2} = 15.5 \text{ buildup for air}$$

$$I = I_o B e^{-\mu\chi}$$

$$I = (0.67)(1.87)(15.5)e^{-(0.894+7.26)}$$

$I = 5.58E-03$ mR/yr at site boundary

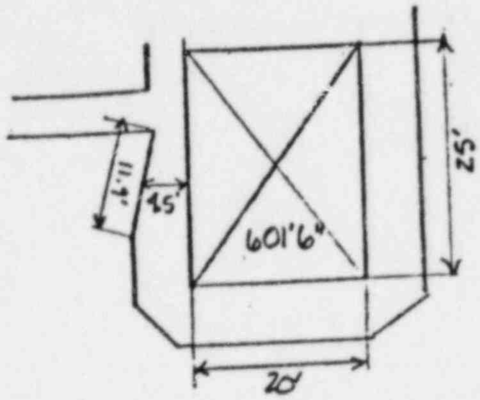
Now we will recalculate the dose using a disk source:

References: Radiation Dosimetry Hine & Brownell Page 763
BRP Drawing M-101
BRP Drawing M-103 Section B

Scaled drawing to find dimensions of fuel pool.

Height of pool 31'

Height of fuel 6'



Equation for disk source where b = o

$$R_\gamma = \frac{q\Gamma}{a^2} \ln \left[\frac{h^2 + a^2}{h^2}\right]$$

where: q = millicuries
Γ = gamma constant
a = radius of disk
b = perpendicular distance of dose point from axis of disk
h = vertical distance of dose point from plane of disk

There are two individual cases to calculate. In both cases it is assumed that the channel rack is stored near the thinnest portion of the spent fuel pool wall and that fuel is stored next to the channel rack which has a concrete wall thickness greater than 4.5 feet.

Case 1 Fuel stored with 1 year decay

Determine radius of disk - height of fuel is 6'
width of spent fuel rack showing through >4.5' concrete thickness is 6.25'

6' X 6.26' = 37.5 sq ft
An equivalent circle πr^2

$\frac{37.5}{\pi} = r^2$ 11.9 = r^2 3.45' = r

Dose rate = 2.5 mR/hr from NUS Safety Analysis for BRP spent fuel case
Graph of dose rate as a function of concrete thickness

h = 5.25' or 160.0cm from RAE calculation is greatest distance between source and dose point and will therefore be most conservative

a = 3.45' or 105.2cm radius of disk

Solve for qΓ

$$q\Gamma = \frac{(R_\gamma)(a^2)}{\ln\left[\frac{h^2 + a^2}{h^2}\right]}$$

$$q\Gamma = \frac{(2.5)(105.2^2)}{\ln \dfrac{160^2 + 105.2^2}{160^2}}$$

$q\Gamma$ = 7.70E+04

Solve for dose rate at the site boundary - approximately 2900 feet or 8.839E+04cm utilizing the shielding effects of the containment shell and air.

Determine buildup factors and μx's assume 0.5 Me$\bar{V}\bar{E}$

containment shell is 3/4" iron

cm = 1.91
ρ = 7.86 g/cm^3
μ/ρ = 0.0840 cm^2/gm
μx = 1.3
B_{iron} = 2.3

2900' of air
34,800" of air

cm = 88,393
ρ = 0.001293 g/cm^3
μ/ρ = 0.0870 cm^2/gm
μx = 9.9

to find buildup for air will use average of buildups for aluminum and water

$B_{aluminum}$ = 38.9

B_{water} = 77.6

$$\frac{38.9 + 77.6}{2} = 58.3 \text{ buildup for air}$$

$$R\gamma = \frac{q\Gamma}{a^2} \ln \left[\frac{h^2 + a^2}{h^2}\right] (Be^{-\mu x})$$

$$R\gamma = \frac{7.70E{+}04}{105.2^2} \ln\left[\frac{88{,}393^2 + 105.2^2}{88{,}393^2}\right] (2.3)(58.3)\, e^{-(1.3+9.9)}$$

$R\gamma$ = 1.807E-08 mR/hr X 8760 hrs/yr

= 1.58E-04 mR/yr @ site boundary

Case 2 Newly irradiated fuel stored

Radius of disk is the same as in Case 1 of 3.45'

Dose rate = ratio dose rates of fuel decayed one year of 38 mR/hr at thinnest portion of wall and 2.5 mR/hr at thicker portion of wall and newly irradiated fuel at thinnest portion of wall at 2300 mR/hr Dose rate obtained from NUS Safety Analysis for Spent Fuel Pool.

$$\frac{38}{2.5} = \frac{2300}{x}$$

Dose Rat = 151.2 mR/hr

Need to solve for $q\Gamma$ Where: $h = 5.25'$ or 160.0 cm
$a = 3.45'$ or 105.2 cm

$$q\Gamma = \frac{(R\ )(a^2)}{\ln\left[\frac{h^2 + a^2}{h^2}\right]}$$

$$q\Gamma = \frac{(151.2)(105.2^2)}{\ln\left[\frac{160^2 + 105.2^2}{160^2}\right]}$$

$q\Gamma = 4.66E+06$

Solve for dose rate at site boundary - approximately 2900 ft or 8.839E+4cm utilizing shielding of containment shield and air

Determine buildup factors and μx's assume 1 Mev$\overline{E}$

Containment shell 3/4" iron

cm = 1.91
$\rho = 7.86\ g/cm^3$
$\mu/\rho = 0.0599$
$\mu x = 0.899$
$B_{iron} = 1.87$

Assume $\mu x = 1$ for buildup

2900' of air
34,800 inches of air

cm = 88,393
$\rho = 0.001293\ g/cm^3$
$\mu/\rho = 0.0636\ cm^2/g$
$\mu x = 7.26$

To find buildup for air will use average of buildups for aluminum and water

$B_{aluminum} = 13.8$ $\qquad \frac{13.8 + 17.1}{2} = 15.5$

$B_{water} = 17.1$

$$R_\gamma = \frac{q\Gamma}{a^2} \ln\left[\frac{h^2 + a^2}{h^2}\right] (Be^{-\mu x})$$

$$R_\gamma = \frac{4.66E+06}{105.2^2} \ln\left[\frac{88,393^2 + 105.2^2}{88,393^2}\right] (1.87)(15.5)e^{-(0.899+7.26)}$$

$R_\gamma = 4.947E-06$ mR/hr X 8760 hrs/yr

$R_\gamma = 4.33E-02$ mR/yr @ site boundary

In doing these two cases there was no consideration made for the decay of the fuel over the year time period.

This first year, the newly irradiated fuel will drop significantly in dose rate in an exponential fashion and then level off. (Almost in half).

To be conservative, we will divide the dose rate obtained for the two cases by two.

Dose rate for newly irradiated fuel = 4.33E-02 mR/yr

Dose rate for fuel decayed 1 year = 1.58E-04 mR/yr

4.33E-02 + 1.58E-04/2 = 2.17E-02 mR/yr average dose rate for newly irradiated fuel

The average dose rate for 1 yr old fuel will be conservatively assumed to remain constant at 1.58E-04 mR/yr

CHRISTA-MARIA CONTENTION 2
AND
O'NEILL CONTENTION IIA

A. THE CONTENTIONS

Christa-Maria Contention 2:

The increase in fuel stored in the Big Rock pool will result in an increase in the amount of radiation released to the environment at the south wall of the storage pool where there is less shielding, according to the licensee's Description and Safety Analysis. This increment in the level of radiation released to the environment enhances the risks to the health and safety of the public in the vicinity of the plant.

O'Neill Contention IIA:

The routine releases of radioactivity during the installation of new racks, the loading of those racks, and storage of fuel in the racks will exceed the exposure of workers, as will the releases of radioactivity through the south wall of the pool exceed the limits imposed by Appendix I to CFR Part 50 on exposure to the general public.

B. MATERIAL FACTS AS TO WHICH THERE IS NO GENUINE ISSUE TO BE HEARD.

1. The south wall of the spent fuel pool tapers from a maximum thickness of 5 ft. 9 inches to a minimum thickness of 3-1/2 feet. (Affidavit of Charles Axtell at p.4).

2. If freshly discharged spent fuel were to be stored adjacent to the 3-1/2 foot thick portion of the south wall of the spent fuel pool, the dose on the outside of the spent fuel pool wall would be 2300 millirem/hr. (Affidavit of William Bell at p.4).

3. If spent fuel which has a decay time of one year or more were to be stored adjacent to the 3-1/2 foot thick portion of the south wall of the spent fuel pool, the dose on the outside of the spent fuel pool wall would be 37.3 millirem/hr. (Affidavit of William Bell at p.4).

4. The dose rate due to radioactive shine from stored spent fuel at all other locations outside the spent fuel pool walls and floor will be less than 2.0 millirem/hr., following the proposed installation of new storage racks. (Affidavit of William Bell, Ex. 1).

5. Following the proposed rack replacement, the existing channel rack will not be moved from its present location at the western (thinnest) portion of the south wall of the spent fuel pool. (Affidavit of Charles Axtell, at p.8 and Attachment C; Affidavit of Roger Sinderman Concerning Radiation Doses To The Public From Fuel Stored Near The South Wall of The Spent Fuel Pool at p.2).

6. It is impossible to store spent fuel in the channel rack. (Affidavit of Charles Axtell at pp.7-8)

7. Therefore, the NUS calculations of 2300 mrem/hr. and 37.3 mrem/hr. are an overestimate of actual dose rates which could occur at the outside of the

south wall of the spent fuel pool due to radioactive shine from spent fuel stored in the Big Rock Point pool following the proposed license amendments. (Affidavit of William Bell at p.3; Affidavit of Charles Axtell at pp. 7-9).

8. Licensee will store only spent fuel with a decay time of one year or more in the outer three rows of the proposed 9 x 9 fuel rack near the south wall of the spent fuel pool. (Affidavit of Charles Axtell at pp. 8-9; Affidavit of Roger Sinderman concerning Radiation Doses to the Public From Fuel Stored Near the South Wall of the Spent Fuel Pcol at p.2).

9. Therefore, the dose rates outside the south wall of the spent fuel pool due to radioactive shine from 1 year old stored spent fuel will be approximately 2 mrem/hr. (Affidavit of Charles Axtell at p.8).

10. A dose rate outside the south wall of 2 mrem/hr. due to radioactive shine from stored spent fuel is small in comparison with the existing dose rate of about 30-40 mrem/hr. due to the filter sock tank.

11. The area outside the tapered portion of the south wall of the spent fuel pool is a radiologically

controlled area which is infrequently entered by plant workmen. (Affidavit of Charles Axtell at p.8).

12. An incremental dose rate of approximately 2 mrem/hr in the infrequently entered, radiologically controlled area outside the south wall of the spent fuel pool would not present a radiation hazard to plant workers or result in occupational exposure exceeding 10 CFR Part 20 limits. (Affidavit of Charles Axtell at pp. 9-11, 17-20).

13. The annual dose rate at the site boundary due to radioactive shine of one year old spent fuel through the south wall of the spent fuel pool would be .00016 millirem/hr. (Affidavit of Roger Sinderman Concerning Radiation Doses to the Public from Fuel Stored Near the South Wall of the Spent Fuel Pool at p.4 and Exhibit 2).

14. This dose rate was conservatively calculated since the attenuation due to forest growth and the auxiliary building structure were neglected. (Affidavit of Roger Sinderman Concerning Radiation Doses to the Public from Fuel Stored Near the South Wall of the Spent Fuel Pool at p.2).

15. A dose rate to the public at the site boundary of .00016 millirem/yr. is within the legal limit

imposed by 10 CFR Part 20 and within 10 CFR Part 50 Appendix I, and therefore radioactive shine through the south wall of the spent fuel pool would not present a risk to the public health and safety. (Affidavit of Roger Sinderman Concerning Radiation Doses to the Public from Fuel Stored Near the South Wall of the Spent Fuel Pool, at pp.4-5).

16. The total radiation exposure to plant workers during the proposed rack replacement operation will be about 18.2 man-rem. (Affidavit of Charles Axtell at pp. 11-16).

17. After installation of the new storage racks, the storage of additional spent fuel will not increase dose rates in the spent fuel pool area. (Affidavit of Charles Axtell at p. 19; NRC Safety Evaluation at pp. 3-13 and 3-14).

18. Refueling operations, which will not be affected by the installation of new storage racks, are the principle cause of occupational exposure to workers from the spent fuel pool since little worker activity occurs over or near the spent fuel pool during routine operations. (Affidavit of Charles Axtell at p.19).

19. Occupational Exposure will not increase due to storage of additional spent fuel in the Big Rock Point spent fuel pool. (Affidavit of Charles Axtell at p.19; NRC Safety Evaluation Report at pp. 3-13 and 3-14).

20. The radiation protection procedures to be used at Big Rock Point Plant for the proposed rack replacement operation and thereafter are in compliance with all applicable federal regulations. (Affidavit of Charles Axtell at p.19).

21. Individual occupational doses during the rack replacement operation and thereafter will be maintained below 10 CFR Part 20 limits by these radiation protection procedures. (Affidavit of Charles Axtell at pp. 17-19).

22. Individual occupational exposure in the vicinity of the spent fuel pool during the proposed rack replacement operation and thereafter will be maintained as low as reasonably achievable. (Affidavit of Charles Axtell at pp. 12-14, 17-19 and Attachment D; NRC Safety Evaluation Report at pp. 3-12 through 3-14).

C. DISCUSSION

The first part of Mr. Axtell's affidavit serves as an introduction to and description of the Big Rock Point spent fuel pool.

The south wall of the spent fuel pool at the level of the stored spent fuel tapers to a minimum thickness of 3-1/2 feet. Christa-Maria Contention 2 and O'Neill Contention IIA express concern about potential exposure to plant workers and to the public due to radioactive shine through the south wall. The affidavits of William Bell and Charles Axtell demonstrate that, taking into account the location of the proposed storage racks and licensee's commitment to store only spent fuel which has decayed at least one year in the outer three rows of the spent fuel rack nearest the south wall, the dose on the outside of the south wall will be about 2 millirem/hr. Moreover, this dose rate will occur in an infrequently visited area which is already radiologically controlled due to the presence of the filter sock tank (30-40 mrem/hr.). Therefore, the incremental dose rate of 2 mrem/hr. in this area due to radioactive shine through the south wall presents no hazard to plant personnel and is as low as reasonably achievable.

The affidavit of Roger Sinderman establishes that the dose rate to the public at the site boundary due to radioactive shine through the south wall, conservatively calculated, is .00016 mrem/hr., which of course is trivial and well within regulatory limits.

The affidavit of Charles Axtell explains the radiation procedures in effect at Big Rock Point Plant, and the specific measures which will be taken during and after the proposed rack replacement operation to ensure that exposure to plant workers is within regulatory limits and as low as reasonably achievable.

These affidavits conclusively establish that radiation from spent fuel through the south wall of the spent fuel pool is not a problem. With respect to potential occupational exposure to plant workers (other than due to shine through the south wall) due to the proposed rack replacement operation and thereafter, Mr. Axtell's affidavit and the NRC Staff's Safety Evaluation Report both describe the measures which will be taken to maintain exposures within regulatory limits and as low as reasonably achievable. Both Mr. Axtell and the NRC Staff conclude that following rack replacement storage of additional spent fuel will not result in additional occupational exposure. Therefore, there is no genuine issue of material fact even with respect to those aspects of Christa-Maria Contention 2 and O'Neill Contention IIA which do not involve the south wall. Summary disposition of both contentions is appropriate.

STATE OF MICHIGAN)
) SS,
COUNTY OF Charlevoix)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CONSUMERS POWER COMPANY	)	Docket No. 50-155
	)	(Spent Fuel Pool Expansion)
(Big Rock Point Nuclear Plant	)	

AFFIDAVIT OF CHARLES E. AXTELL

I, Charles E. Axtell, of lawful age, being first duly sworn, do state as follows:

I am employed by Consumers Power Company as the Plant Health Physicist at the Big Rock Point Plant. I have held this job for 13 years. In this job my responsibilities include monitoring and controlling personnel exposure, ALARA considerations, controlling off-site releases of radioactive materials, and plant water chemistry. My resume has been previously submitted.

I am the author of the Testimony of Charles E. Axtell concerning Christa-Maria's Contention 2 and O'Neill Contention 11.2.

To the best of my knowledge and belief, the statements in this affidavit and in the above testimony and Figures and Attachments thereto are true and correct.

Charles E. Axtell

Charles E. Axtell

SUBSCRIBED AND SWORN TO
before me this 2nd day of
October, 1981

Patricia E. Kujawski

Notary Public

PATRICIA E. KUJAWSKI
Notary Public, Charlevoix County, Mich.

Eugene A. Dziedzic
March 6, 1983

PATRICIA E. KUJAWSKI
Notary Public, Charlevoix County, Mich.
My Commission ...