Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

September 23, 1985
LIC-85-420

Mr. E. J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. Letter OPPD (W. C. Jones) to NRC (D. G. Eisenhut) dated January 26, 1981.
3. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August, 1974.
4. San Onofre 2 and 3, FSAR, Appendix 15B, Dose Models Used to Evaluate the Environmental Consequences of Accidents.

Dear Mr. Butcher:

NUREG-0737, Item III.D.3.4
Control Room Habitability

In accordance with NUREG-0737, Item III.D.3.4, Omaha Public Power District submitted a study which confirmed that 1) control room operating personnel would be adequately protected in the event of accidental releases of radionuclides or toxic gases, and 2) the plant could be safely operated or shut down in the event of design basis accidents.

The results of the study indicated that operator radiation exposure for a postulated LOCA were less than the limits established by the Standard Review Plan (SRP) 6.4 for beta skin and gamma whole body doses. However, the thyroid dose exceeded the SRP 6.4 limits. As a result, OPPD took corrective actions by installing an iodine monitor in the Control Room in order to monitor iodine exposure during a LOCA. In addition, a shield wall was installed in the Control Room corridor to reduce the direct shine radiation from containment spray piping.

The corrective actions and the study, (Reference 2) were reviewed and found to be acceptable by the NRC. However, during the latter part of 1984, the NRC performed similar calculations using the Campe and Murphy methodology and the assumptions provided with the Reference 2 study. Because the NRC results reflected a higher whole body gamma dose, questions were raised concerning the X/Q dispersion values. Subsequently, OPPD reviewed the calculations supporting the study and concluded that the initial analysis was acceptable and the

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results were conservative due to the diffused source model and virtual distance assumptions used. The NRC reviewers agreed that the calculations were conservative. However, with no experimental data available to prove the virtual distance theory, the calculations were considered to be unacceptable.

In order to resolve the control room habitability issue, and to further reduce post accident radiation doses to the control room operators, OPPD will extend the control room air intake (VA-65) from the existing location (3.9 meters from the containment wall) to a distance of at least 10 meters from the containment wall. The new calculated doses using a 10 meter distance and NRC accepted methodology are lower than the limits identified in SRP 6.4.

A summary of the control room habitability calculated doses are included in Attachment A. Justification for continued operation (including input data for the computer calculations, assumptions and the resulting doses), until shutdown for the scheduled refueling outage are included in Attachment B. These doses represent the control room air intake in its present location of 3.9 meters from the containment wall. Attachment C represents the input data for the computer calculation, assumptions and resulting doses for the control room air intake relocated to 10 meters from the containment wall. Checking and design verification of these calculations is currently being performed and will be finalized prior to plant restart.

The additional calculations performed and the proposed relocation of the VA-65 ductwork provide substantial justification to complete resolution of the control room habitability issue. The relocation of the intake will be completed prior to restart after the Fall 1985 refueling outage.

Sincerely,

R. L. Andrews

R. L. Andrews
Division Manager
Nuclear Production

RLA/AB/rs

Attachments

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
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Mr. E. G. Tourigny, NRC Project Manager
Mr. L. A. Yandell, NRC Senior Resident Inspector

ATTACHMENT A

Summary of
Control Room Habitability
Calculated Doses

Additional calculations have been performed for the existing configuration (3.9 meters duct distance) in order to justify continued operation until shutdown for the 1985 refueling outage. These calculations were done using the Campe and Murphy methodology, (Reference 3) and plant specific assumptions. The results of the calculations and assumptions used are included in Attachment B. The following doses were calculated:

Dose (thyroid) = 12.81 rem < 30 rem(1)

Dose (Gamma Whole Body) = 2.7 rem < 5 rem(1)

Dose (Beta Skin) = 25.0 rem < 30 rem(1)

To resolve the Control Room habitability issue, the control room filtered air intake (VA-65) will be extended from its existing location (3.9 meters from the containment wall) to at least 10 meters from the containment wall. These calculations were done using the Campe and Murphy methodology, Reference 3. The results of the calculations and assumptions used are included in Attachment C. The following doses were calculated:

Dose (thyroid) = 10.97 rem < 30 rem(1)

Dose (Gamma Whole Body) = 2.83 rem < 5 rem(1)

Dose (Beta Skin) = 26.99 rem < 30 rem(1)

Based on the above doses the operator exposures are below SRP 6.4 limits.

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(1) NUREG-0800, Standard Review Plan (Section 6.4)

ATTACHMENT B

Justification for Continued Operation (3.9 meter location)

I. Assumptions

The following specific assumptions were used in the analysis.

1. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at 100% of the ultimate core power level of 1500 Mwt.

2. One hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from the containment.

3. Twenty-five percent of the core equilibrium radioactive iodine inventory is immediately available for leakage from the containment.

4. The release of fission products to the containment is assumed to occur instantaneously and transported instantaneously to the environment and Control Room assuming no decay during transport time.

5. Of the iodine fission product inventory released to the containment, 91% is in the form of elemental iodine, 5% is in the form of particulate iodine, and 4% is in the form of organic iodine.

6. Radioactive Decay - Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident to the Control Room.

7. Containment Iodine Removal System - Credit for the removal of iodine from the containment building atmosphere is assumed during the course of the accident resulting from filtration in the iodine removal system. The system consists of four air handling units; two having filtering capacity and the other two having no filtering capacity. In the calculation of the radiological consequences an adsorption efficiency of 0.9 (90%) and operation of only one of the two containment filtering units is assumed. Credit for elemental, particulate, and organic iodine removal is taken for the course of the accident.

8. The following removal constants for the containment iodine removal systems are assumed in the analysis: (Ref. USAR Section 14.15.4)

Elemental Iodine	-	5.14 hr^{-1}
Organic Iodine	-	5.14 hr^{-1}
Particulate Iodine	-	5.14 hr^{-1}

9. The containment is assumed to leak at 0.05 vol.%/d for the duration of the accident. This is conservative since under actual conditions the containment pressure decreases with time. By reducing the leak rate as a function of containment pressure, the operator exposures will be less than calculated. The test results from 1983 integrated leak rate testing indicate 0.047%/day at 95% confidence level.

10. The radioactive leakage from containment is assumed to be uniform from the entire surface area of containment building. Therefore, the probability of leakage is the same per unit surface area from containment walls or from containment dome. Three major leakage pathways have been identified as the following:

    (a) The leakage from containment walls into the adjacent building, such as the spent fuel storage area and the remaining portions of Auxiliary Building, which will be eventually exhausted to the ventilation stack.

    (b) The leakage from containment dome to the outside environment.

    (c) The leakage from containment walls to the outside environment. Refer to Figure 1 for release pathways (a), (b), and (c).

The release pathway (a) and (b) will not increase the radionuclide concentrations at the fresh air intake since no downwash can occur at 1 m/sec wind velocity and due to 60 ft of elevation difference with the control room intake. However, in order to be conservative, pathway (b) from containment dome is assumed to be released as a ground level release at the same elevation as the control room intake.

Under accident conditions the release pathway (a), which is leakage into the Auxiliary Building, is most probable since the Electrical and Mechanical penetrations are located in the Auxiliary Building and since any releases to the Auxiliary Building would be eventually exhausted to the ventilation stack. The only exceptions are the releases from the containment wall into the switchgear room, cable spreading room, lower electrical, compressor room and room 81 which exhaust to VA-41 at the same elevation as the Control Room intake. However since VA-41 is downwind from the Control Room intake it will not contribute to the radionuclide concentrations at the intake. Therefore, assuming release from pathways (b) and (c) is extremely conservative.

Additionally, the release at all the elevations from pathway (c) is assumed to occur at the same elevation as the fresh air intake (ground level release and ground level receptor). This is also an extremely conservative assumption since the releases from elevations up to 60 feet above the fresh air intake cannot be observed at the elevation of the fresh air intake due to the fact that no significant downwash can occur at 1 m/sec wind velocity at close distance of 12 ft to the containment where the intake is located. Higher wind velocities, e.g., 10m/sec which may result in downwash were considered. However, X/Q was also reduced by a factor of 10 since X/Q and wind velocity are inversely proportional. Therefore, even with downwash, the operator exposures were lower than the results shown in this report.

11. The X/Q dispersion model is assumed to be a diffused source model from Campe and Murphy (Ref. 3) with standard deviations in vertical crosswind (σ_z) and horizontal crosswind (σy) set equal to zero. This is extremely conservative, since there is significant turbulence within the building wake boundary.

12. All the activity released during H_2 purge operation is assumed to be initiated 4 days after LOCA up to 30 days after LOCA. This is conservative since the Fort Calhoun Updated Safety Analysis Report indicates initiation of H_2 purge 5 days after LOCA up to 22 days after LOCA. The contribution from H_2 purge operation is negligible with the Auxiliary Building release flow rate of 72,500 CFM due to high exit velocity and no significant downwash from pathway (a). As indicated above, downwash can occur at 10 m/sec wind velocity. However since X/Q decreases by a factor of 10 at 10 m/sec, and the location of the intake is 12 ft from containment wall and 60 ft below the ventilation exhaust, the contribution from H_2 purge will be negligible.

13. The adjustment factors stated in Table 1 of Campe and Murphy (Ref. 3) for Control Room occupancy have been modified as shown below:

TABLE 1

FACTORS USED TO CALCULATE EFFECTIVE

RELATIVE CONCENTRATIONS FOR SELECTED TIME INTERVALS

Adjustment Factors	0 - 8 hrs	8 - 24 hrs	1- 4 days	4 - 30 days
Occupancy	1	0.25	0.50	0.40
Wind Speed	1	0.67	0.50	0.33
Wind Direction	1	0.88	0.75	0.50
Overall Reduction	1	0.147	0.1875	0.066

The occupancy factors for Control Room operators is based on work schedule shown on Table 2.

TABLE 2

OPERATOR WORK SCHEDULE

	0 - 8 hrs	*8 - 24 hrs	1 - 4 days	4 - 30 days
Hours in Control Room	8 hrs	4 hrs	12 hrs/day	--
Occupancy Factor	1.0 (From Table 1)	0.25	0.5	0.4 (From Table 1)

* It is assumed that an operator spent a total of 12 hours per day during the first 4 days of the accident in the Control Room.

14. A semi-infinite cloud equation for beta dose calculation was used from Campe and Murphy. This equation is conservative since the Control Room volume is finite. Beta exposure can be further reduced if credit is taken for attenuation of beta through the suspended ceiling at 9' above the floor level.

II. Input Parameters for Computer Calculations

LOCA II computer code from Combustion Engineering is used to calculate the operator Gamma, Beta and Iodine radiation exposures. The equations used in the code are shown and discussed in SAN ONOFRE 2 & 3 FSAR Appendix 15B (Reference 4).

. Input Parameters for LOCA II are as Follows:

(1) Iodine Removal Constant

5.14 hr^{-1} from Fort Calhoun USAR (Section 14.15.4)

(2) X/Q Atmospheric Dispersion Factor

The equation from Campe and Murphy (Ref. 3) for diffused source is used:

$$X/Q = [U(\pi \cdot \sigma y \cdot \sigma z + \frac{a}{K+2})]^{-1}$$

Where;

X/Q = Relative Concentration Dispersion Factor (sec/m^3)
σy, σz = Standard deviation of the gas concentration in horizontal crosswind and vertical crosswind respectively (m)
U = Wind velocity = 1 m/sec
a = cross sectional area of containment building = 1340 m^2 from Fort Calhoun USAR

$$K = \frac{3}{(s/d)^{1.4}} = \frac{3}{(3.9/36)^{1.4}} = 67.37$$

s = Distance between containment and control room intake = 3.9 meter
d = Diameter of containment = 36 meter

$$X/Q = [1 \text{ m/sec } (\pi \cdot (0) \cdot (0) + \frac{1340}{67.37 + 2})]^{-1}$$

$X/Q = 5.18 \times 10^{-2}$ sec/m^3
(0-8 hr)

Multiplying the X/Q for 0-8 hours by reduction factors from Table 1 the X/Q for other time intervals is obtained.

$X/Q = 5.18 \times 10^{-2} \times 0.147 = 7.61 \times 10^{-3}$ sec/m^3
(8-24 hr)

$X/Q = 5.18 \times 10^{-2} \times 0.1875 = 9.71 \times 10^{-3}$ sec/m^3
(1-4 days)

$X/Q = 5.18 \times 10^{-2} \times 0.066 = 3.41 \times 10^{-3}$ sec/m^3
(4-30 days)

(3) Containment Leak Rates

The leak rates from containment are as follows:

Containment Time Interval	% Leak Rate in 24 hr Period
0-8 hr	0.05%
8-24 hr	0.05%
1-4 days	0.05%
4-30 days	0.05%

(a) Leakage Area Adjacent to Auxiliary Building

Leakage pathway (a) as discussed in the assumptions:

The total surface area of containment enclosed in the Auxiliary Building will be calculated as follows:

as shown on Drawing 11405-A-8 (attached) the containment has been divided in four major segments;

- 130° Segment with Roof Elevation of 1057'

  $$\text{Arc length} = \frac{130°}{360°} \times \pi \times D$$

  $$= \frac{130°}{360°} \times \pi \times 116' = 131.6 \text{ ft}$$

  Height of 130° segment = El. 1057' - El.995' = 62 ft

  Area of 130° segment = 62' x 131.6' = 8159 ft^2

- 112° Segment with Roof Elevation of 1045'

  $$\text{Arc length} = \frac{112°}{360°} \times \pi \times D = 113.4 \text{ ft}$$

  Height of segment = El. 1045' - El. 995' = 50 ft

  Area of 130° segment = 50' x 113.4' = 5670 ft^2

- 36° Segment with Roof Elevation of 1083'

  $$\text{Arc length} = \frac{36°}{360°} \times \pi \times D = 36.4 \text{ ft}$$

  Height of segment = El. 1083' - El. 995' = 88 ft

  Area of segment = 88' x 36.4'= 3203 ft^2

- <u>27° Segment with Roof Elevation of 1035'-8</u>

  Arc length $= \frac{27°}{360°} \times \pi \times D = 27.3$ ft

  Height of the segment = El. 1035'.8 - El. 995' = 40.8 ft

  Area of the segment = 40.8' x 27.3' = 1114 ft^2

Total surface area of containment wall enclosed in Auxiliary Building = 130° segment area + 112° segment area + 36° segment area + 27° segment area.

$= 8159 + 5670 + 3203 + 1114 = 18146\ ft^2$

(b) <u>Leakage Area From Containment Dome</u>

Leakage pathway (b) as discussed in the assumptions:

Containment dome Area (circular approximation) $= \frac{(116)^2\pi}{4} = 10568\ ft^2$

(c) <u>Leakage Area From Containment Walls to the Outside Environment</u>

Leakage pathway (c) as discussed in the assumptions:

To calculate the surface area of the containment walls leaking to the outside environment; total surface area of containment wall is calculated and then subtraced by the area of containment enclosed in the Auxiliary Building:

Total surface area = (El. 1119'-El. 995') π D = 45189 ft^2

Surface area of containment wall leaking to outside environment = 45189 - 18146 = 27043 ft^2

As stated in the assumptions, the leakage from ventilation stack pathway (a) will not contribute to the concentration of radionuclides at the control room intake since no downwash can occur due to the following reasons:

The release from Auxiliary Building ventilation stack is assumed at 72500 CFM. Since no downwash can occur if $\frac{w}{u} \geq 5$; (from Reg. Guide 1.111)

Where u = wind velocity

w = release velocity = (72500 CFM)/34.9 ft^2 cross sectional area of stack

= 2077 ft/min or 10.55 m/sec

$\frac{10.55 \text{ m/sec}}{1 \text{ m/sec}} \geq 5$ thus no downwash can occur

In addition the ventilation stack exhaust is 60 feet higher in elevation than the Control Room intake and due to close proximity of the intake to the containment wall, even if downwash were to occur it would not affect the intake which is 12 ft away from containment.

Therefore the only section of containment leakage which may contribute to the radionuclide concentration at the Control Room intake is the release from walls of containment exposed to the outside environment (pathway c) and release from containment dome (pathway b). As stated earlier the release from the containment walls at different elevations is all assumed to be condensed and released at the same elevation as the Control Room intake. The total leakage rate from pathway (c) and (b) are calculated as follows:

total leakage from containment is assumed at 0.05% for the duration of the accident;

$$\text{Containment Leak Rate} \times \text{Containment Free Volume} = 0.0005/\text{day} \times 1.05 \times 10^6 \ ft^3 = 5.25 \times 10^2 \ ft^3/\text{day}$$

$$5.25 \times 10^2 \ ft^3/\text{day} \div 24 \ \text{hr/day} = 2.1875 \times 10^1 \ ft^3/\text{hr}$$

The leakage for the containment walls and dome exposed to the environment assuming uniform release from all surface areas is calculated as follows:

$$\text{Leak Rate From Containment Wall Exposed to Outside} = \text{Total Containment Leak Rate} \times \frac{\text{Containment Wall Area Exposed to Outside \& Containment Dome Area}}{\text{Total Containment Surface Area (Wall \& Dome)}}$$

$$= (2.1875 \times 10^1\ ft^3/hr) \times \frac{27043\ ft^2 + 10568\ ft^2}{45189\ ft^2 + 10568\ ft^2}$$

$$= 1.475 \times 10^1\ ft^3/hr$$

$$\text{Fraction of Leak Rate} = \frac{\text{Containment Leak Rate}}{\text{Containment Free Volume}} = \frac{1.475 \times 10^1\ ft^3/hr}{1.05 \times 10^6\ ft^3} = 1.405 \times 10^{-5}\ hr^{-1}$$

The above leak rate will be assumed for time intervals 0-8 hr, 8-24 hr, 1-4 day, and 4-30 days.

(4) Radionuclide Activities in Containment at Time Zero After LOCA

The source term activies are based on release of 100% of core noble gases and 25% of core iodines to the containment atmosphere:

NUCLIDE	ACTIVITY (Ci)
KR-85	0.33×10^6
KR-85m	0.11×10^8
KR-87	0.19×10^8
KR-88	0.28×10^8
XE-131m	0.29×10^6
XE-133	0.85×10^8
XE-135	0.15×10^8
XE-135m	0.17×10^8
XE-138	0.68×10^8
I-131	0.10×10^8
I-132	0.15×10^8
I-133	0.21×10^8
I-134	0.23×10^8
I-135	0.20×10^8

(5) Breathing Rates

The breathing rates are based on values specified by Reg. Guide 1.4.

(6) The Control Room Free Volume

100,000 ft^3

This value represents the actual dimensions of the Control Room space. USAR value of 45,100 represents free volume of the Control Room below the suspended ceilings.

(7) Control Room Infiltration Flow

1100 CFM x 60 min/hr = 66,000 ft^3/hr

(8) Build Up Rate for Radionuclide Concentration

$$\text{Build Up Rate} = \frac{66000\ ft^3\ /hr}{100,000\ ft^3} = 0.66\ hr^{-1}$$

(9) The Semi-Infinite Cloud Approximation for Gamma Dose and Beta Dose

$D_\gamma\ \ semi_\infty = 0.25 \cdot (X/Q) \cdot \sum_i Qi\ Ei$ from Campe and Murphy

A geometry factor for a finite volume from Campe and Murphy is applied

$$GF = \frac{1173}{\underset{(V)}{0.338}} = \frac{1173}{(100,000\ ft^3)^{0.338}} = 23.95$$

$$D_\gamma = \frac{0.25 \cdot (X/Q) \cdot \sum_i Qi\ Ei}{23.95}$$

$$D_\beta\ Semi_\infty = 0.23 \cdot (X/Q) \cdot \sum_i Qi\ Ei$$

III. OPERATOR EXPOSURES

(1) The integrated iodine thyroid dose to the Control Room operators for the duration of LOCA from airborne activity in Control room is

D_I = 12.81 Rem

The gamma whole body and beta skin integrated radiation dose to the Control Room operators for duration of LOCA from airborne activity in Control Room are:

D_γ = 2.1 Rem < 5 Rem (SRP 6.4 limit)
D_β = 25.0 Rem < 30 Rem (SRP 6.4 limit)

NOTE: The actual Beta dose calculated in the air is 60 rem, however, the ratio of $\frac{\text{Beta skin}}{\text{Beta air}}$ factors from Reg. Guide 1.109 were applied. Taking credit for Beta attenuation through the outer dead layer of skin is stated in Reg. Guide 1.109. Refer to attached output sheet from LOCA II B Program for conversion of beta air dose to beta skin dose.

(2) The operator gamma whole body exposure from radioactive cloud over the Control Room roof for duration of the LOCA is:

Whole Body < 0.1 REM

(3) The operator gamma whole body exposures from iodine filter in Room 81 and other sources i.e. containment spray piping and containment building for duration of LOCA is:

Whole Body < 0.5 REM

Total Operator Exposures From all Radioactive Sources:

D_γ = 2.7 REM
D_β = 25.6 REM
D_I = 12.81 REM

All the exposures are below SRP.6.4 limits.

IV. CONCLUSIONS

The operator integrated dose for whole body gamma, beta skin, and iodine thyroid calculated for duration of LOCA are less than the limits of Standard Review Plan Section 6.4. The dose calculations are extremely conservative as stated in various sections of this report. Major conservatisms which will further reduce the above doses are:

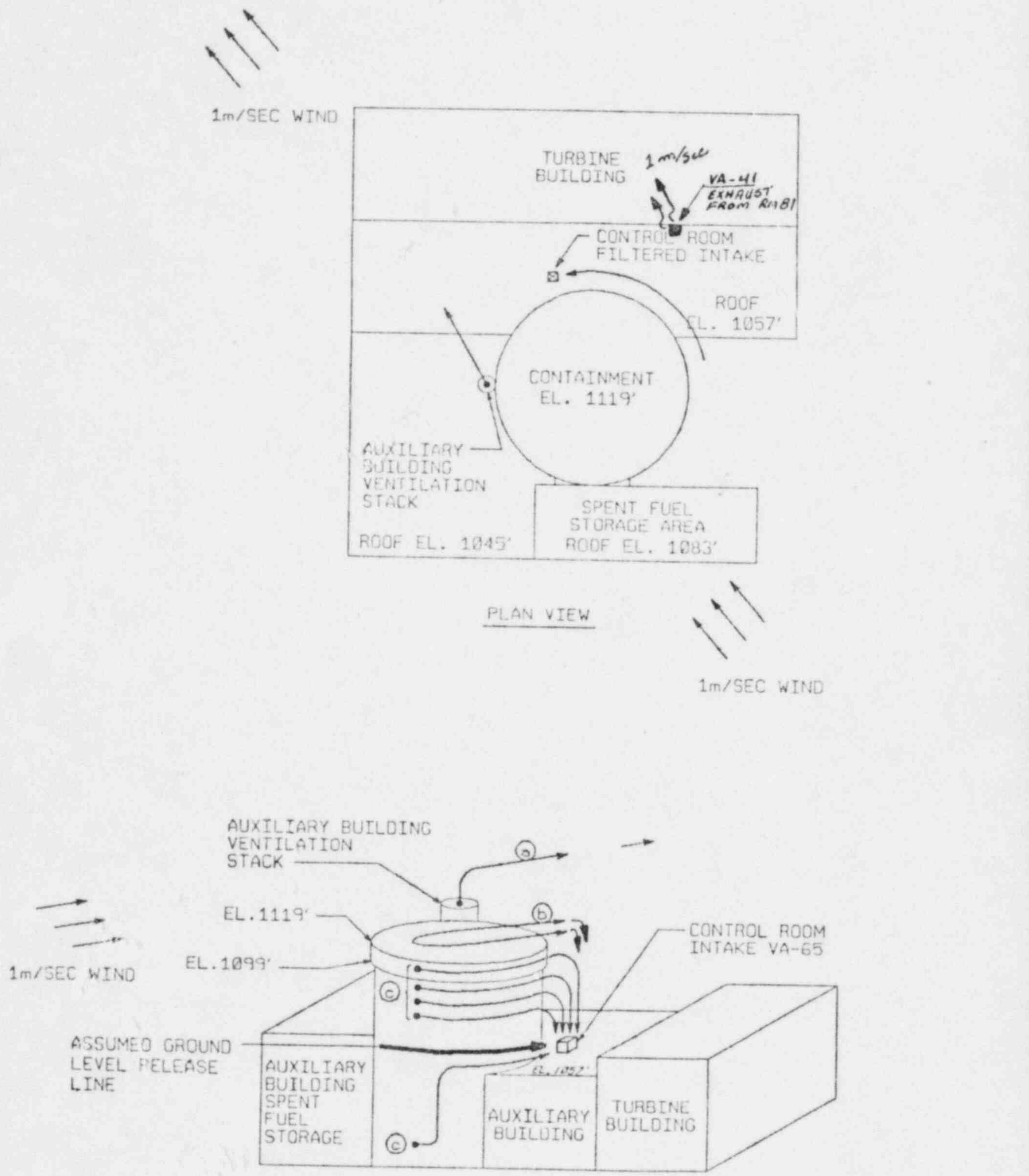
(1) The containment leak rate tests at Fort Calhoun indicate leak rate of 0.047%/day. Reduction in leak rate as a function of post LOCA containment pressure can further reduce the operator exposures.

(2) The actual σy and σz values are larger than zero assumed in this report which can decrease the X/Q value which will in turn reduce operator exposures.

(3) The releases from pathway (c) or the exposed walls of containment are assumed to be ground level releases at the same elevation of the Control Room intake. Taking credit for release at different elevations would further reduce the calculated dose by a factor of two or more.

Therefore, the calculated doses are believed to be a factor of two or more higher than actual if a more realistic set of assumptions were used.

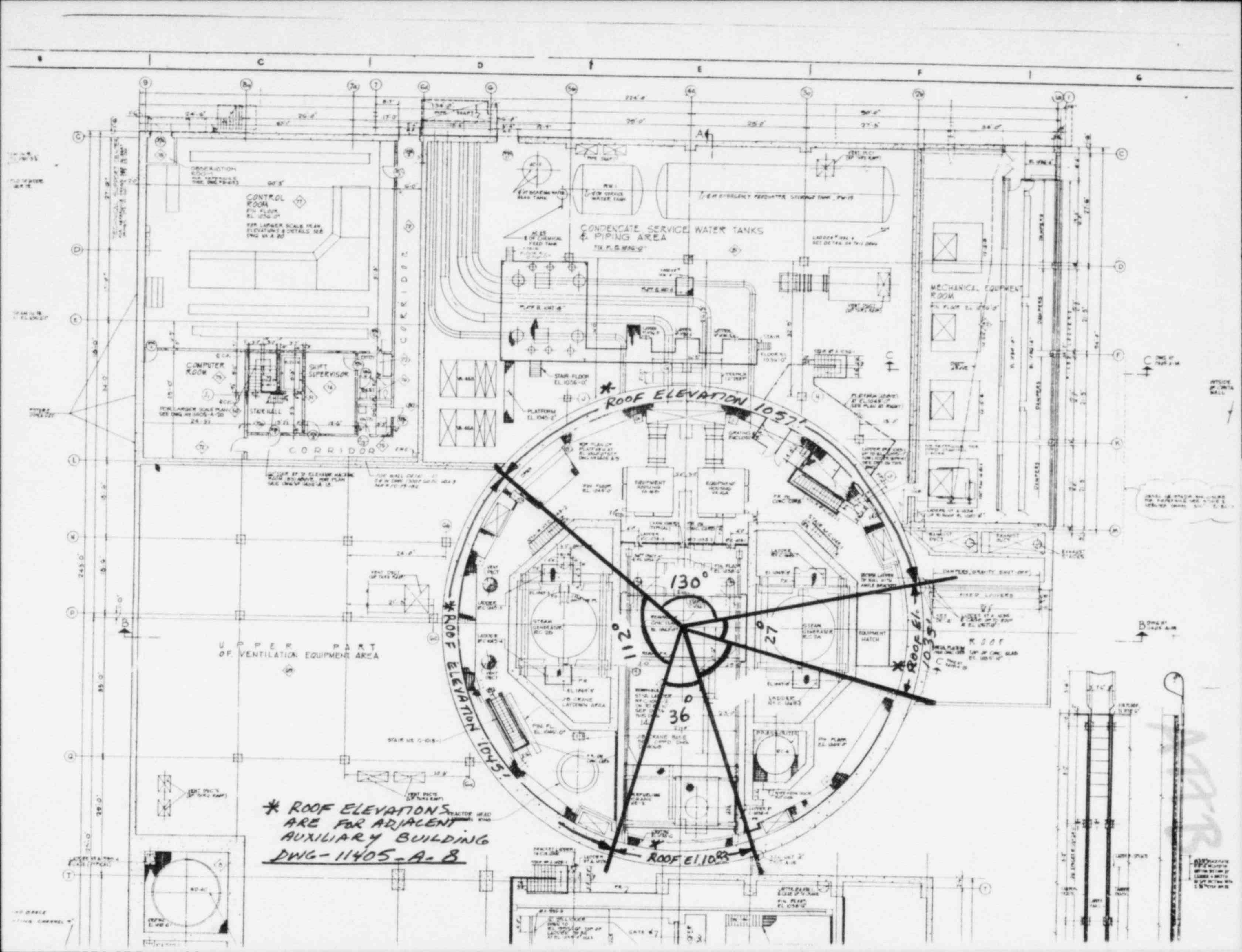
FIGURE 1



ⓐ THE LEAKAGE TO ADJACENT AUXILIARY BUILDING AND EVENTUAL RELEASE FROM THE VENTILATION STACK.

ⓑ THE LEAKAGE FROM CONTAINMENT DOME HAS ALSO BEEN ASSUMED TO BE RELEASED AT ASSUMED GROUND LEVEL RELEASE LINE SHOWN ABOVE.

ⓒ THE LEAKAGE FROM EXPOSED PORTIONS OF CONTAINMENT WALL. THE PATHWAY "c" RELEASES AT DIFFERENT ELEVATIONS HAVE ALL BEEN CONDENSED AND RELEASED AT THE ASSUMED GROUND LEVEL LINE SHOWN ABOVE.



CONTROL ROOM
COMPUTER ROOM
SHIFT SUPERVISOR
STAIR HALL
CORRIDOR
CONDENSATE, SERVICE WATER TANKS & PIPING AREA
MECHANICAL EQUIPMENT ROOM
UPPER PART OF VENTILATION EQUIPMENT AREA
ROOF
* ROOF ELEVATION 1057'
* ROOF ELEVATION 1045'
* ROOF El. 1035'
* ROOF El. 1083
130°
112°
27°
36°
* ROOF ELEVATIONS ARE FOR ADJACENT AUXILIARY BUILDING
DWG-11405-A-8

LOCA II·B. OUTPUT. P.

NUCLIDE	GAMMA RAYS	BETA RAYS (MEV/DIS)
KR-85	0.0021	0.228
KR-85M	0.1560	0.354
KR-87	0.8560	1.014
KR-88	2.0000	0.307
XE-131M	0.0220	0.135
XE-133	0.0455	0.126
XE-135	0.2480	0.310
XE-135M	0.4400	0.090
XE-138	0.9320	0.565
I-131	0.3750	0.209
I-132	2.2900	0.421
I-133	0.6360	0.403
I-134	2.5080	0.558
I-135	1.4570	0.475

SITE BOUNDARY WHOLE BODY DOSES DUE TO

* FROM PAGE 1.109-21 OF REG. GUIDE 1.109; BETA SKIN & BETA AIR FACTORS (mREM-m³/pCi-YR) & (mRAD-m³/pCi-YR)

$(\beta_{AIR}) \times \frac{\beta_{SKIN}}{\beta_{AIR}}{}^{*}$ = BETA DOSE IN DEAD LAYER OF SKIN (REM)

NUCLIDE	GAMMA RAYS TIME=1	GAMMA RAYS TIME=4	BETA RAYS TIME=1	BETA RAYS TIME=4	
KR-85	3.47E-05	3.37E-04	8.30E-02	8.06E-01	× 1.34(-3) / 1.95(-3) = → 0.55
KR-85M	4.16E-02	4.49E-02	2.08E+00	2.25E+00	→ × 1.46(-3) / 1.97(-3) = → 1.66
KR-87	1.23E-01	1.23E-01	3.21E+00	3.22E+00	× 9.73(-3) / 1.03(-2) = → 3.04
KR-88	1.00E+00	1.04E+00	3.39E+00	3.51E+00	→ × 2.37(-3) / 2.93(-3) = → 2.84
XE-131M	3.17E-04	1.83E-03	4.28E-02	2.48E-01	→ NEGLEGIBLE
XE-133	1.86E-01	7.37E-01	1.13E+01	4.49E+01	→ × 3.06(-4) / 1.05(-3) = → 13.08
XE-135	1.33E-01	1.65E-01	3.65E+00	4.55E+00	× 1.86(-3) / 2.46(-3) = → 3.44
XE-135M	4.32E-03	4.32E-03	1.95E-02	1.95E-02	
XE-138	3.01E-02	3.01E-02	4.02E-01	4.02E-01	
I-131	5.72E-05	5.72E-05	7.02E-04	7.03E-04	
I-132	3.37E-04	3.37E-04	1.36E-03	1.36E-03	→ NEGLEGIBLE.
I-133	1.88E-04	1.88E-04	2.62E-03	2.62E-03	
I-134	3.33E-04	3.33E-04	1.63E-03	1.63E-03	
I-135	3.58E-04	3.58E-04	2.57E-03	2.57E-03	
		2.1 γ RAD		60 β RAD IN AIR	TOTAL 24.6 ≈ 25 REM BETA SKIN DOSE

NUCLIDE	TOTAL OF GAMMA AND BETA TIME = 1	TIME = 4	
KR-85	8.31E-02	8.07E-01	
KR-85M	2.12E+00	2.29E+00	
KR-87	3.33E+00	3.34E+00	
KR-88	4.39E+00	4.54E+00	
XE-131M	4.31E-02	2.50E-01	
XE-133	1.15E+01	4.57E+01	
XE-135	3.79E+00	4.72E+00	
XE-135M	2.38E-02	2.38E-02	
XE-138	4.32E-01	4.32E-01	
I-131	7.59E-04	7.60E-04	
I-132	1.70E-03	1.70E-03	
I-133	2.81E-03	2.81E-03	
I-134	1.97E-03	1.97E-03	
I-135	2.93E-03	2.93E-03	
TOTAL	2.57E+01	6.21E+01	RADS

γ DOSE = 2.1 RAD (REM)
β SKIN DOSE = 25 REM

MESSAGE SUMMARY: MESSAGE NUMBER - COUNT

208 63

ATT 5

ATTACHMENT C

Calculations for Relocation of the Fresh Air
Intake to 10 meters

I. Assumptions

The following assumptions were used in calculating the operator exposures by relocating the filtered air (VA-65) intake to 10 meters away from containment wall:

1. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at 100% of the ultimate core power level of 1500 Mwt.

2. One hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from the containment.

3. Twenty-five percent of the core equilibrium radioactive iodine inventory is immediately available for leakage from the containment.

4. The release of fission products to the containment is assumed to occur instantaneously and transported instantaneously to the environment and Control Room assuming no decay during transport time.

5. Of the iodine fission product inventory released to the containment, 91% is in the form of elemental iodine, 5% is in the form of particulate iodine, and 4% is in the form of organic iodine.

6. Radioactive Decay - Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident to the Control Room.

7. Containment Iodine Removal System - Credit for the removal of iodine from the containment building atmosphere is assumed during the course of the accident resulting from filtration in the iodine removal system. The system consists of four air handling units; two having filtering capacity and the other two having no filtering capacity. In the calculation of the radiological consequences an adsorption efficiency of 0.9 (90%) and operation of only one of the two containment filtering units is assumed. Credit for elemental, particulate, and organic iodine removal is taken for the course of the accident.

8. The following removal constants for the containment iodine removal systems are assumed in the analysis: (Ref. USAR Section 14.15.4)

Elemental Iodine	-	5.14 hr^{-1}
Organic Iodine	-	5.14 hr^{-1}
Particulate Iodine	-	5.14 hr^{-1}

9. The containment is assumed to leak at 0.1 vol.%/d for the duration of the accident per Fort Calhoun Technical Specification. This is conservative since under actual conditions the containment pressure decreases with time. By reducing the leak rate as a function of containment pressure, the operator exposures will be less than calculated. The test results from 1983 integrated leak rate testing indicate 0.047%/day at 95% confidence level.

10. All the leakage from containment walls is assumed to be a ground level release and ground level receptor at the same elevation as the control room filtered intake.

11. The X/Q dispersion model is assumed to be a diffused source model from Campe and Murphy (Ref. 3) with standard deviations in vertical crosswind (σ_z) and horizontal cross wind (σy) set equal to zero. This is extremely conservative, since there is significant turbulence within the building wake boundary. The wind velocity was assumed at 1 m/sec for calculations X/Q.

12. The contribution from H_2 purge operation is negligible from ventilation stack, with the Auxiliary Building release flow rate of 72,500 CFM due to high exit velocity and no downwash. Downwash can occur at 10 m/sec wind velocity, however since X/Q decreases by a factor of 10 at 10 m/sec, and the location of the intake is 12 ft from containment wall and 60 ft below the ventilation stack exhaust, the contribution from H_2 purge will be negligible.

13. The control room filtered intake flow rate was assumed at 1200 CFM.

14. The adjustment factors stated in Table 1 of Campe and Murphy (Ref. 3) for Control Room occupancy have been used to adjust X/Q values.

15. A semi-infinite cloud equation for beta dose calculation was used from Campe and Murphy. This equation is conservative since the Control Room volume is finite. Beta exposure can be further reduced if credit is taken for attenuation of beta through the suspended ceiling at 9' above the floor level.

II. Calculations

LOCA II computer code from Combustion Engineering is used to calculate the operator, Gamma, Beta and Iodine radiation exposures. The equations used in the code are shown and discussed in the SAN ONOFRE 2 & 3 FSAR Appendix 15B (Ref. 4).

- Input Parameters for LOCA II are as Follows:

(1) Iodine Removal Constant

5.14 hr^{-1} from Fort Calhoun USAR (Section 14.15.4)

(2) X/Q Atmospheric Dispersion Factor

The equation from Campe and Murphy (Ref. 3) for diffused source is used:

$$X/Q = [U(\pi \cdot \sigma y \cdot \sigma z + \frac{a}{K+2})]^{-1}$$

Where;

X/Q = Relative Concentration Dispersion Factor (sec/m^3)
σ y, σ z = Standard deviation of the gas concentration in horizontal crosswind and vertical crosswind respectively (m)
U = Wind velocity = 1 m/sec
a = cross sectional area of containment building = 1340 m^2 from Fort Calhoun USAR

$$K = \frac{3}{(s/d)^{1.4}} = \frac{3}{(10/36)^{1.4}} = 18.03$$

S = Distance between containment and control room intake = 10 meter
d = Diameter of containment = 36 meter

$$X/Q = [1 \text{ m/sec } (\pi \cdot (0) \cdot (0) + \frac{1340}{18.03 + 2})]^{-1}$$

$X/Q = 1.49 \times 10^{-2}$ sec/m^3
(0-8 hr)

Multiplying the X/Q for 0-8 hours by reduction factors from Table 1 of Campe and Murphy, will provide the X/Q's for other time intervals;

$X/Q = 1.49 \times 10^{-2} \times 0.59 = 8.8 \times 10^{-3}$ sec/m^3
(8-24 hr)

$X/Q = 1.49 \times 10^{-2} \times 0.23 = 3.4 \times 10^{-3}$ sec/m^3
(1-4 days)

$X/Q = 1.49 \times 10^{-2} \times 0.066 = 9.9 \times 10^{-3}$ sec/m^3
(4-30 days)

(3) Containment Leak Rates

The leak rates from containment are as follows:

Containment Time Interval	% Leak Rate in 24 hr Period
0-8 hr	0.1%
8-24 hr	0.1%
1-4 days	0.1%
4-30 days	0.1%

NOTE: 0.1% is convervative since actual integrated leak rate test at Fort Calhoun indicated 0.047% at 95% confidence level during 1983 testing.

Fraction of Leak Rate for Containment $= \frac{.001/\text{day}}{24\ \text{hrs/day}} = 4.17 \times 10^{-5}\ \text{hr}^{-1}$

(4) Radionuclide Activities in Containment at Time Zero After LOCA

The source term activies are based on release of 100% of core noble gases and 25% of core iodines to the containment atmosphere:

NUCLIDE	ACTIVITY (Ci)
KR-85	0.33×10^6
KR-85m	0.11×10^8
KR-87	0.19×10^8
KR-88	0.28×10^8
XE-131m	0.29×10^6
XE-133	0.85×10^8
XE-135	0.15×10^8
XE-135m	0.17×10^8
XE-138	0.68×10^8
I-131	0.10×10^8
I-132	0.15×10^8
I-133	0.21×10^8
I-134	0.23×10^8
I-135	0.20×10^8

(5) Breathing Rates

The breathing rates are based on values specified by Reg. Guide 1.4.

(6) The Control Room Free Volume

$100{,}000\ ft^3$

(7) Control Room Infiltration Flow

$1200\ CFM \times 60\ min/hr = 72{,}000\ ft^3/hr$

(8) Build Up Rate for Radionuclide Concentration

$$\text{Build Up Rate} = \frac{72{,}000\ ft^3/hr}{100{,}000\ ft^3} = 0.72\ hr^{-1}$$

(9) The Semi-Infinite Cloud Approximation for Gamma Dose and Beta Dose

$D_\gamma\ semi_\infty = 0.25 \cdot (X/Q) \cdot \sum_i Qi\ Ei$ from Campe and Murphy

A geometry factor for a finite volume from Campe and Murphy is applied

$$GF = \frac{1173}{(V)^{0.338}} = \frac{1173}{(100{,}000\ ft^3)^{0.338}} = 23.95$$

$$D_\gamma = \frac{0.25 \cdot (X/Q) \cdot}{23.95}\ Qi\ Ei$$

$D_\beta\ Semi_\infty = 0.23 \cdot (X/Q) \cdot \sum_i Qi\ Ei$

OPERATOR EXPOSURES

(1) The integrated iodine thyroid dose to the Control Room operators for the duration of LOCA from airborne activity in Control room is

$D_I = 10.97$ Rem < 30 rem (SRP 6.4 limit)

The gamma whole body and beta skin integrated radiation dose to the Control Room operators for duration of LOCA from airborne activity in Control Room are:

D_γ = 2.23 Rem < 5 Rem (SRP.6.4 limit)
D_β = 26.99 Rem < 30 Rem (SRP.6.4 limit)

NOTE: The actual Beta dose calculated in the air is 67.3 rem, however, the ratio of $\frac{\text{Beta skin}}{\text{Beta air}}$ factors from Reg. Guide 1.109 were applied. Taking credit for Beta attenuation through the outer dead layer of skin is stated in Reg. Guide 1.109.

(2) The operator gamma whole body exposure from radioactive cloud over the Control Room roof for duration of the LOCA is:

Whole Body < 0.1 REM

(3) The operator gamma whole body exposures from iodine filter in Room 81 and other sources i.e. containment spray piping and containment building for duration of LOCA is:

Whole Body < 0.5 REM

Total Operator Exposures From all Radioactive Sources:

D_γ = 2.83 REM
D_β = 26.99 REM
D_I = 10.97 REM

All the exposures are below SRP.6.4 limits.

III. CONCLUSIONS

The operator integrated dose for whole body gamma, beta skin, and iodine thyroid calculated for duration of LOCA are less than the limits of Standard Review Plan Section 6.4. The dose calculations are extremely conservative as stated in various sections of this report. Major conservatisms which will further reduce the above doses are:

(1) The containment leak rate tests at Fort Calhoun indicate leak rate of 0.047%/day, however by using the conservative value of 0.1%/day the operator doses calculated above are increased by more than a factor of 2. Reduction in leak rate as a function of post LOCA containment pressure can further reduce the operator exposures.

(2) The actual σy and σz values are larger than zero assumed in this report which can decrease the X/Q value which will in turn reduce operator exposures.

(3) It is extremely conservative and unrealistic to assume that all activity from containment walls will be released at the same elevation as the fresh air intake through a solid concrete wall (ground level release and receptor).

Therefore, the calculated doses are believed to be a factor of four or more higher than actual if a more realistic set of assumptions were used as it has been demonstrated in Attachment B for justification for continued operation.

There is a strong need to establish standards which will allow realistic release pathways for radionuclides from containment walls during a LOCA in order to obtain a more realistic operator exposures.

IV. CORRECTIVE ACTIONS

In order to increase the margin of safety for operator exposures during LOCA, control room filtered intake (VA-65) will be relocated to at least 10 meters away from containment wall. In addition, an iodine monitor was installed in Control Room for continuous monitoring of post LOCA iodine in the Control Room.