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Director of Nuclear Reactor Regulation
Att: Mr Albert Schwencer, Chief
Operating Reactor Branch No 1
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - FUEL HANDLING
ACCIDENT IN CONTAINMENT

Our March 22, 1977 submittal relating to the consequences of a refueling accident inside the containment of the Palisades Plant contained typographical errors in Appendices B, C and D.

The entire report is resubmitted at this time and previous copies should be destroyed.

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CC: JGKeppler, USNRC

REFUELING ACCIDENT INSIDE CONTAINMENT FOR THE PALISADES PLANT

A. SOURCE TERM

The worst refueling accident that could occur inside containment is defined as the hottest bundle dropping onto the cavity floor at two days after reactor shutdown with the immediate consequency of one outer row of fuel rods breaking.[1] The radioactive material released is as listed in Table 1. Appendix A gives all assumptions made in deriving Table 1.

TABLE 1

SOURCE TERM (2 DAYS AFTER SHUTDOWN)

Isotope	Activity (Ci)
I-131	3.48E01
I-133	1.81E01
Xe-133	7.13E03
Xe-133m	5.02E03
Xe-135	2.46E02
Kr-85	9.78E02

B. PUFF RELEASE

Assuming the activities listed in Table 1 are released simultaneously, the Area Radiation Monitors will alarm immediately and initiate containment isolation valve closure. Since the time duration for closing the isolation valve is less than the transit time for the radioactive puff to reach the exhaust grill, release of radioactive material through the containment stack does not occur. Appendix B gives detailed description of the puff release.

C. UNIFORM MIXING OF ACTIVITY RELEASED WITH CONTAINMENT AIR

If the release from the cavity is gradual and the activity uniformly mixed with containment air, the exposure rate at the two ARMs is sufficient to signal containment isolation using semi-infinite cloud model. The amount of activity released from the stack prior to isolation is less than the case analyzed in (D). Details are in Appendix C.

D. WITHOUT ISOLATION

Assuming the release rate from the cavity does not cause containment isolation and consequently all the activity is released through the stack, the whole body and thyroid doses to individuals at site boundary during the entire period of release are 29 mRems and 6.2 rems respectively.

This case is not believed credible, however, it does represent the upper limit of the consequence of a refueling accident. Since these consequences are far less than Part 100 limits, no detailed dose consequences are necessary for Case (C) which is a more realistic evaluation of a probable accident description.

NOTES:

(1) FSAR, Section 14.19.2.

APPENDIX A - SOURCE TERM

Assumptions:

1. Reactor has been in operation at 2,650 MW_t for 3 full years prior to shutdown.

2. Reg Guide 1.25 assumptions for the radial peaking factor, gap activity, effective decontamination factor for water are adapted.

3. Accident occurs at 48 hours after shutdown (FSAR 14.19.2).

4. Thirteen fuel rods were broken (FSAR 14.19.2).

5. Fission yields from U-235, and half-lives of radionuclides, are adapted from Battelle Northwest Laboratory's Chart of the Nuclides.

Isotope	Y(%)	$T_{1/2}$	df	Activity (Ci)
I-131	2.9	8.05 d	100	3.48 E01
I-133	6.5	20.3 h	100	1.81 E01
Xe-133	6.5	5.27 d	1	7.13 E03
Xe-133m	6.5	2.26 d	1	5.02 E03
Xe-135	6.4	9.2 h	1	2.46 E02
Kr-85	1.3	10.76 y	1	9.78 E02

APPENDIX B - PUFF RELEASE

Area Radiation Monitor

There are two area radiation monitors (one-out-of-two logic for isolation) inside containment during refueling period. The set point to initiate containment isolation valve closure is 20 mR/hour above background for both ARMs. Maximum distance from the cavity to either ARM is less than 9 meters.

Exposure Rate at ARMs

Isotope	Activity (Ci)	τ (R/hr/Ci @ 1 m)	$1/D^2$ (9 m)	Exposure Rate (R/h)
I-131	3.48 E01	.22	.012	9.18 E-02
I-133	1.81 E01	.294	.012	6.38 E-03
Xe-133	7.13 E03	.01	.012	8.56 E-01
Xe-133m	5.02 E03	.01	.012	6.02 E-01
Xe-135	2.46 E02	.14	.012	4.27 E-01
Kr-85	9.78 E02	.004	.012	4.69 E-02
Total				2.03 R/h

The exposure rate at the ARMs is 2.03 R/h from the puff released to the water surface. This exposure rate is sufficient to signal IV closure.

Transit Time for the Puff From the Cavity Surface to the Exhaust Grill

The distance between the cavity surface to the exhaust grill is 43 feet. The air movement speed inside containment is 3 feet/second. Therefore, it takes 14.3 seconds for the puff to get to the exhaust grill assuming the air is pushing at that direction.

IV Closure Time

The Tech Specs require routine closure testing of this valve and plant procedure QO-5 limits the closure time to 10 seconds or less. A 4-second closure time is routinely experienced.

Conclusion

Since the transit time is greater than both the Tech Spec and typical IV closure time, there will be no radioactive material released from the exhaust duct.

APPENDIX C - UNIFORM MIXING

ARM Response

Assuming uniform mixing with containment air (1.684×10^6 cf or 4.77×10^4 m^3), the exposure rate to the ARMs using semi-infinite cloud model is 8.44 R/hr.

Isotope	Ci	X (Ci/m^3)	$\bar{E}_\gamma$	$\bar{E}_\gamma X$	Exposure Rate (mR/h)
I-131	3.48 E01	7.30 E-04	.38	2.77 E-04	2.49 E02
I-133	1.81 E01	3.79 E-04	.50	1.90 E-04	1.71 E02
Xe-133	7.13 E03	1.49 E-01	.03	4.47 E-03	4.02 E03
Xe-133m	5.02 E03	1.05 E-01	.03	3.15 E-03	2.84 E03
Xe-135	2.46 E02	5.16 E-03	.25	1.29 E-03	1.16 E03
Total					8.44 E03 mR/h

This is sufficient to close the isolation valve.

APPENDIX D - DOSE AT SITE BOUNDARY WITHOUT ISOLATION

A. Thyroid Dose

From CHNG 56-75 (10/17/75) attached thyroid dose to individuals at site boundary during the entire period of radioactive plume traveling is 6.2 rems.

B. Whole Body Dose

Using the semi-infinite cloud model, the whole body dose to individuals at site boundary during the entire period of radioactive plume traveling is 29 mRems.

Isotope	Ci	X/Q	X	$\bar{E}_\gamma$	$\bar{E}_\gamma X$
I-131	3.48 E01	2.6×10^{-4}	9.05 E-03	.38	3.44 E-03
I-133	1.81 E01	2.6×10^{-4}	4.71 E-03	.50	2.35 E-03
Xe-133	7.13 E03	2.6×10^{-4}	1.85	.03	5.56 E-02
Xe-133m	5.02 E03	2.6×10^{-4}	1.31	.03	3.92 E-02
Xe-135	2.46 E02	2.6×10^{-4}	6.40 E-02	.25	1.60 E-02
Total					1.16 E-01

$D = .25\ \Sigma \bar{E}_\gamma X$

$= 2.9E\text{-}02$ Rem = 29 mRem

THYROID DOSE AT SITE BOUNDARY FOR THE WORST FUEL HANDLING ACCIDENT WITHOUT USING THE CHARCOAL FILTER

A. Definition for the Worst Fuel Handling Accident and Limit for Thyroid Dose at Site Boundary:

The worst accident that could occur in the spent fuel pool during fuel handling is defined as the hottest fuel bundle drops onto the spent fuel pool floor and one outer row of fuel rods breaks.[1] The thyroid dose at site Boundary due to this worst accident shall not exceed 1.5 rem, which is the limit for a complete loss-of-load incident.[2] Since the probability of a fuel handling accident is less than the probability of a loss-of-load accident, the assumption is justified.

B. Reactor Core Iodine Inventory:

A simplified formula for the reactor core inventory, q, for a specific isotope is given by equation (1).

$$q_{(Ci)} = \frac{Po(MW_t) \times 3.2 \times 10^{16}\ (\text{fission/sec}/MW_t) \times Y \times (1 - e^{-\frac{.693}{Tr} To})}{3.7 \times 10^{10}\ (\text{dis/sec/Ci})} \quad \ldots(1)$$

Where:

q is the amount of isotope contained by the reactor at shutdown (Ci).

Po is the rated reactor power level (MW_t), 2650.

Y is the fission yield.

Tr is the radiological half life of the isotope.

To is the effective full power time, 3 years.[3]

Values for the reactor core inventory of the iodine isotopes are given in Table 1.

(1) FSAR, Section 14.19.2

(2) Tech. Spec., Section 3.1.4

(3) Assuming the entire core is irradiated for 3 years at full power.

Table 1. Reactor Core Inventory of Iodine Isotopes - At Shutdown

Isotope	Fission Yield (%)	Half-Life	Inventory (Ci)
I-129	.9	1.7×10^7 y	2.52
I-131	2.9	8.05 d	6.65 E 07
I-132	4.4	2.26 h	1.00 E 08
I-133	6.5	20.3 h	1.49 E 08
I-134	7.6	52.0 m	1.74 E 08
I-135	5.9	6.68 h	1.35 E 08

C. Activity Released from the Damaged Fuel Assembly:

The amount of activity released from the damaged fuel assembly under the worst accident is given by equation (2).

$$q'_{(Ci)} = \frac{q_{(Ci)} \times (RPF) \times (Fg) \times (F)}{N} \quad \ldots(2)$$

Where:

q' is the activity released (Ci).

q is defined in paragraph B.

N is the number of fuel assemblies in the core, 204.

RPF is the radial peaking factor, 1.65[4].

Fg is the fraction of fuel in the gap.

F is the fraction of the assembly damaged, 0.077[5].

Values of activities released for the iodine isotopes are given in Table 2.

Table 2. Activity Released from the Damaged Fuel Assembly - At Shutdown

Isotope	Fg	Activity (Ci)
I-129	.3	4.70 E-04
I-131	.1	4.14 E 03
I-132	.1	6.23 E 03

(4) US NRC, Regulatory Guide 1.25.

(5) FSAR, Section 14.19.2.

I-133	.1	9.32 E 03
I-134	.1	1.09 E 04
I-135	.1	8.39 E 03

D. Effective Decontamination Factor for Spent Fuel Pool Water:

The radioactive iodines released is composed of inorganic and organic species. The composition and the decontamination factors of these two species are given in Table 3.

Table 3. Composition and Decontamination Factors for the Iodine Species[6]

Species	Composition	DF
Inorganic	99.75%	133
Organic	.25%	1

The effective decontamination factor for spent fuel pool water is:

$$EDF = \frac{1}{.9975/133 + .0025/1} = 100$$

E. Activity Inhaled at Site Boundary:

The activity inhaled at site boundary by a "Standard Man"[7] is given by equation (3).

$$R\ (Ci) = \frac{q'\ (Ci) \times (x/Q) \times B}{EDF} \quad ...(3)$$

Where:

R is the activity inhaled at site boundary (Ci).

q' is defined in paragraph C.

x/Q is the dispersion coefficient at site boundary, 2.6×10^{-4} sec/m^3.

B is the breathing rate of a "Standard Man" 3.47×10^{-4} m^3/sec.

EDF is defined in paragraph D.

(6) US NRC, Regulatory Guide 1.25.

(7) ICRP Publication No 2.

Values of activity inhaled at site boundary for the different iodine isotopes are given in Table 4.

Table 4. Activity Inhaled at Site Boundary - At Shutdown

Isotope	Activity Inhaled (Ci)
I-129	4.24 E-13
I-131	3.73 E-06
I-132	5.62 E-06
I-133	8.39 E-06
I-134	9.78 E-06
I-135	7.57 E-06

F. Thyroid Dose at Site Boundary:

The dose to the thyroid gland of a "Standard Man" due to the iodine isotopes inhaled is given by equation (4).

$$D\ (\text{rem}) = 8.54 \times 10^{2}\ \frac{fa\ \bar{E}\ Te}{M}\ R\ (\text{Ci}) \quad \ldots(4)$$

Where:

D is the thyroid dose (Rem).

fa is the fraction of the amount inhaled gets into the thyroid, 0.23[8].

$\bar{E}$ is the effective energy of the isotope (MeV).

Te is the effective half life of the isotope (sec).

M is the thyroid weight, 20 grams.

R is defined in paragraph E.

Values of the thyroid dose for the different iodine isotopes are given in Table 5.

[8] ICRP Publication No 2.

Table 5. Thyroid Dose at Site Boundary - At Shutdown

Isotope	E (MeV)	Te (days)	Dose (rem)
I-129	0.068	138	3.38 E-06
I-131	0.23	7.6	5.53
I-132	0.65	0.097	3.01 E-01
I-133	0.54	0.87	3.34
I-134	0.82	0.036	2.00 E-01
I-135	0.52	0.28	9.71 E-01

G. Thyroid Dose at Site Boundary as A Function of Post-Removal Time:

The thyroid dose due to I-131 and I-133 as a function of post-removal time from the core is plotted in Figure 1. Since I-129 has a very small thyroid dose, 3.38×10^{-6} rem and I-132, I-134 and I-135 all have very short half-lives, the thyroid dose due to I-131 and I-133 is approximately equivalent to total thyroid dose during post-removal time 5-150 days. From Figure 1, it can be seen that at 16th day post-removal, the thyroid dose from a worst fuel handling accident is less than 1.5 rem and at 69th day post-removal, the thyroid dose is less than 15 m rem. All thyroid dose calculations are done without the operation of the charcoal filter.

H. Conclusion:

Fuel movement without activating the charcoal filter should be allowed if it is more than 16 days since its removal from the core. Should a worst fuel handling accident occur at this time, the thyroid dose at site boundary would be less than 1.5 rem.

CHNG: 56-75

10/17/75

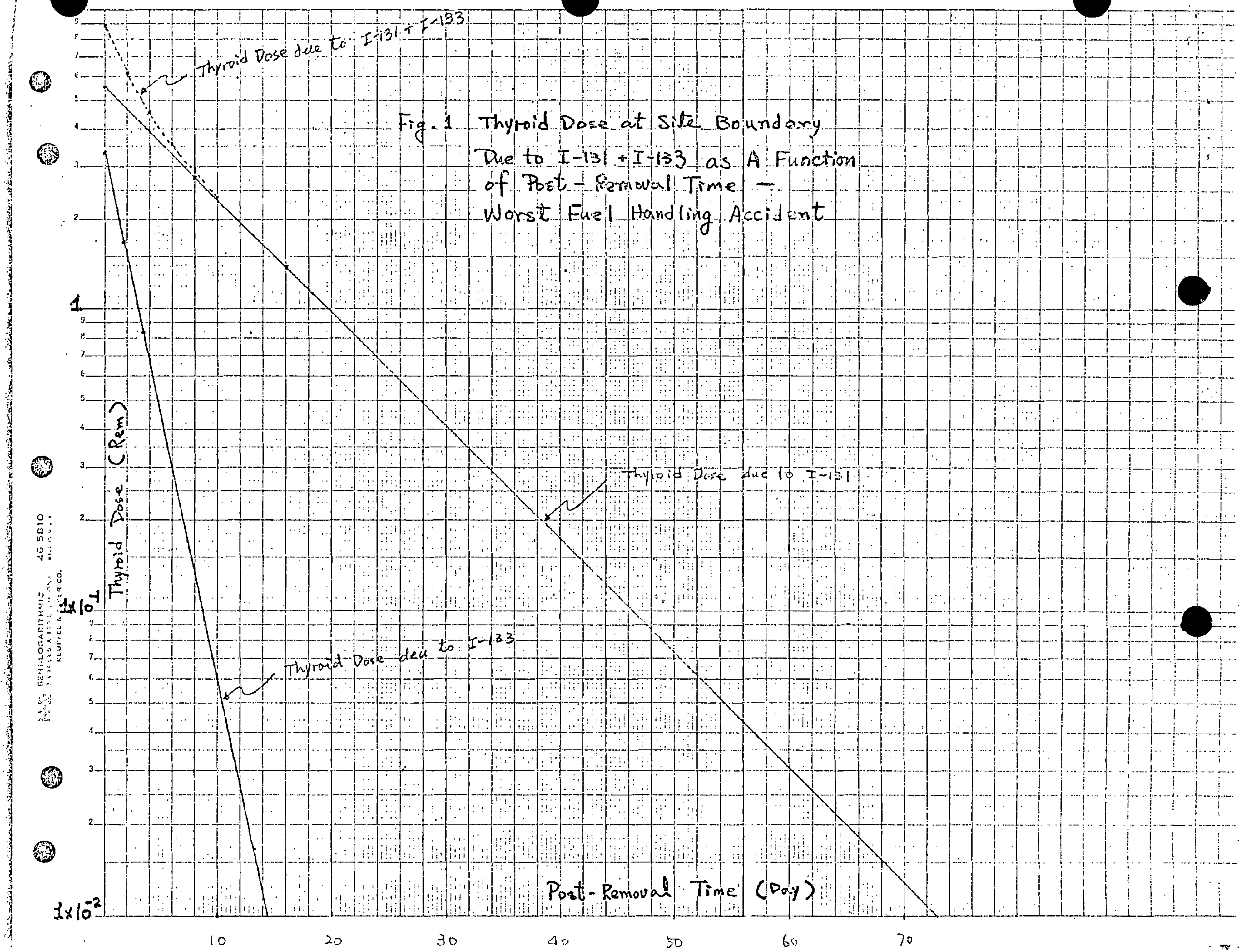

Fig. 1 Thyroid Dose at Site Boundary Due to I-131 + I-133 as A Function of Post-Removal Time — Worst Fuel Handling Accident