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■ 論 文 ■

A Control Room Dose Assessment for a 1300 MWe PWR Following a Loss of Coolant Accident

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Abstract

The habitability of a reactor control room in a French 1300 MWe P'4 type PWR has been evaluated through the exposure dose assessment for the reatcor operator following a Loss of Coolant Accident.

The main hypotheses adopted in this evaluation are based on the French Standard Safety Analysis Report.

A simple computer program, named COREX(COntrol Room EXposure), was developed to calculate : the time-integrated radioactivities released from the reactor building, the volume factors for radionuclides concerned and the resulting time-integrated external whole body and internal thyroid doses to the reactor operators staying in the control room up to 30 days following the LOCA.

The results obtained in this study, on the whole, well agreed with those proposed by the EDF(Electricite de France) except for the case of the whole body exposure, which was attributed to the differences in the volume factors for the radionuclides concerned.

I. Introduction

A dose assessment for the reactor operator after a Loss of Coolant Accident(LOCA) is indispensable for evaluating the habitability of the reactor control room.

The habitability of a reactor control room is assured by the following basic principles[1,4] :

1) A reactor control room should be secured in safe condition in normal operations and even in the design basis accidents like a LOCA.
2) The radiation protection should be properly maintained to the reactor operator during any worst accident situation.

When a LOCA breaks out, the atmosphere inside the reactor containment building is contaminated by fission gass(mainly noble gases) and radioiodines from the reactor core and the primary coolant system. The contaminated air leaks out of the reactor building to reach the atmosphere and ventilation system, results in radiation exposure to a reactor operator[3].

Figure 1 shows a flow chart for a evaluation of the control room habitability and Figure 2 shows a schematic drawing of a French 1300 MWe P'4 type reference RWR building for which the control room habitability is evaluated.

II. Methods of Calculation

II.1 Loss of Coolant Accident

No fission product except for those of gaseous type and organic form of iodine(such as CH_3I) is released into the atmosphere because almost all of the particulates and molecular fission products are removed by the emergency containment spray system at the time of the LOCA[1,4].

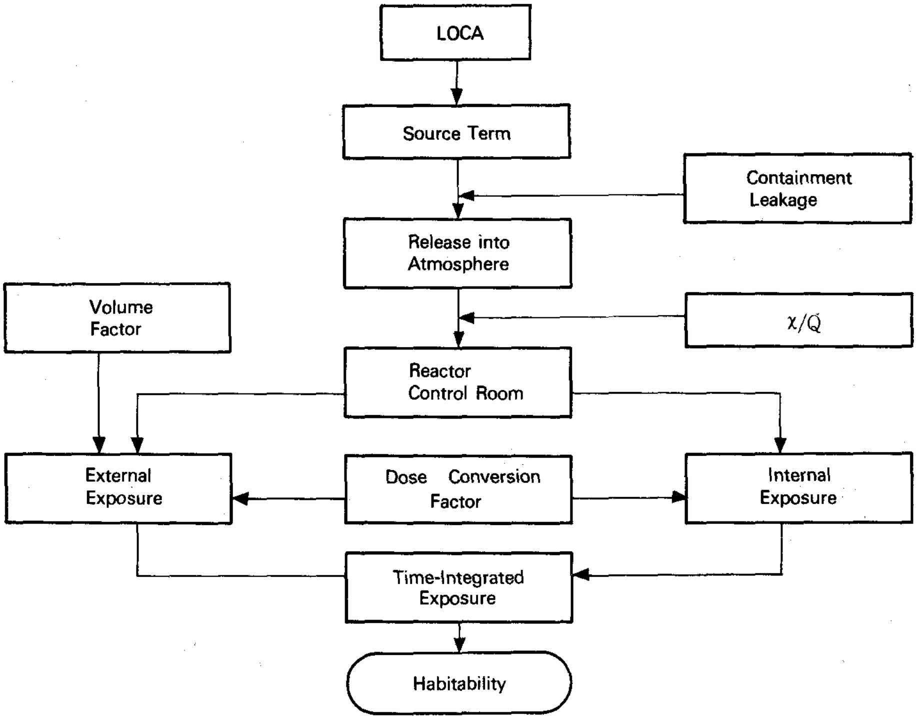

Fig. 1. Flow chart for the control room dose assessment after after a LOCA for the habitability evaluation.

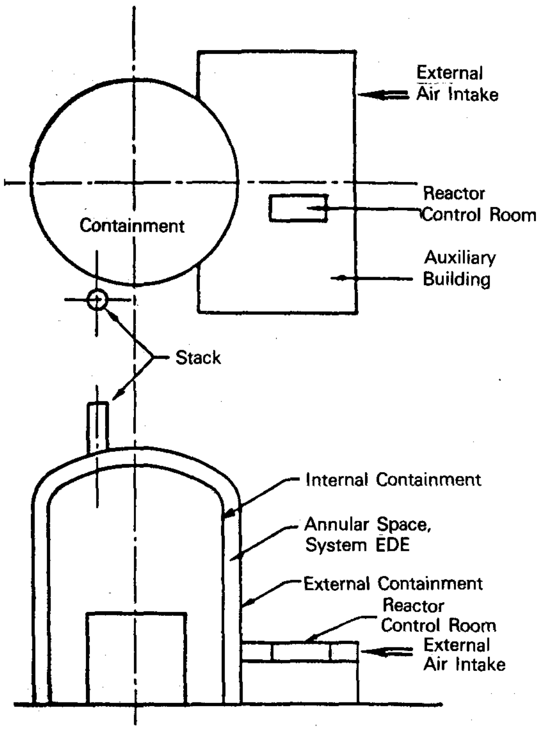

Fig. 2. Schematic drawing of a 1300 MWe PWR containment, reactor control room and its stack.

The parameters used in this accident analysis such as the reactor power, the fuel damage and the fuel irradiation time ar e listed in Table 1. Radioactive source terms in the reactor core and the containment atmosphere at the time of a LOCA are summarized in Table 2.

II.2 Fission Product Release into the Atmosphere

The French 1300 MWe PWR is characterized by its double containment concrete wall(inner-containment, annular space and outer-containment as illustrated in Figure. 2) and two modes of fission product

Table 1. Parameters related to a LOCA and its source term.

Content	Parameters	Remarks
a. LOCA		
−Reactor power	4100 MWth	max. possible,
−Fuel rod rupture	100%	conservatism
−Fuel irradiation time		
· Zone 1	1 year	
· Zone 2	1 year	
· Zone 3	3 years	
b. FP inventory in coolant		
−Noble gases	2.0E-02 of core inventory	Kr-85 : 0.3
−Iodines	3.0E-02 of core inventory	
· Elemental (90%)		I_2 etc.
· Organic (10%)		CH_3I etc.
c. Containment spray removal fraction		Ref. 3
−Noble gases	none	
−Elemental iodines	0.999	
−Organic iodines	none	
d. FP in containment atmosphere		
−Noble gases	2.0E-02 of core inventory	
−Elemental iodines	2.7E-05 of core inventory	
−Organic iodines	3.0E-03 of core inventory	
e. Removal efficiency of filtration system in annular space and in control room		Ref. 1,2
−Noble gases	0.0	No removal
−Radioidines	0.9	90% removal
f. Iodine deposition or redeposition on structure material		

release(due to leakage) from the containment can be assumed to occur[1,2,4]:

A direct and an indirect fission product leakages due to the design-basis consideration, depending on the pressure inside the reactor building during the LOCA.

The fission products radioactivities released from the reactor building due to direct leakage are

Table 2. Fission product inventory in core and in containment atmosphere for a 1300 MWe P'4 type PWR at the time of a LOCA.

Nuclide	Half-life	$E\gamma$(MeV)[1]	Core(10^7Ci)[2]*	Fraction of emission	Containment atmo.(10^4Ci,t=0)
Kr 83cm	1.86h	0.032	1.445	2E-02	28.90
Kr 85cm	4.48h	0.158	3.264	2E-02	65.28
Kr 85	10.71y	0.514(0.437%)	0.075	3E-01	22.50
Kr 87	76.00m	0.70	6.360	2E-02	120.20
Kr 88	2.80h	1.973	8.737	2E-02	174.70
Xe 131m	11.93d	0.019	0.077	2E-02	1.54
Xe 133m	2.23d	0.218	0.657	2E-02	13.14
Xe 133	5.29d	0.044	23.16	2E-02	463.20
Xe 135m	15.60m	0.432	4.268	2E-02	85.36
Xe 135	9.20h	0.250	5.655	2E-02	113.10
Xe 138	14.10m	1.060	19.50	2E-02	390.00
I 131	8.04d	0.390	11.06	3.03E-03	33.51
I 132	2.30h	2.400	16.05	3.03E-03	48.63
I 133	21.00h	0.617	23.10	3.03E-03	69.99
I 134	52.80m	2.632	25.38	3.03E-03	76.90
I 135	6.61h	1.640	21.72	3.03E-03	65.81

1) CEA-R-4284. Rev. 1 (1983)
2) CEA-SERMA-S-314 (1977)
* Core in equilibrium at a final stage of irradiation.

calculated by

$$QD_i(t_a,t_b) = \int_{t_b}^{t_a} Q_{oi} \cdot FD(t_a,t_b) \cdot \exp(-\lambda_i t)\ dt \cdots 1)$$

where $QD_i(t_a, t_b)$ is the released radioactivity of radionuclide i by direct leakage during(t_b-t_a) after LOCA, Q_{oi} is the radioactivity of a radionuclide i existed in the containment atmosphere at time zero(ci), $FD_i(t_a,t_b)$ is a direct leak rate from the containment during(t_b-t_a) after LOCA(mass fraction of fission gases retained in the containment per day,) and λ_i : radioactive decay constant of radionuclide i (h_{-1}).

On the other hand, because of the existence of a filtration system in the annular space, the fission products leaked into this space by indirect leakage are captured by this filtration system before release to the atmosphere through the stack.

The fission products radioactivities released in this stage is given by

$$QI_i(t_a,t_b) = \int_{t_a}^{t_b} Q_{oi} \cdot FI(t_a,t_b) \cdot DR(t_a,t_b) \cdot DF_i \cdot \exp(-\lambda_i t)\, dt \cdots\cdots 2)$$

where $QI_i(t_a, t_b)$ is the released radioactivity of radionuclide i by indirect leakage during($t_b - t_a$) after LOCA (Ci), $FI(t_a, t_b)$ is total indirect leak rate(mass fraction of fission gases retained in the containment per day) which equals (leak fraction from inner-con-

Table 3. Leak rate of the P'4 type PWR containment building (fraction of total mass of gas retained in the containment atmosphere per day).

Time passed after an accident	0h～1h	1h～2h	2h～6h	6h12h	12h～24h	after 24h
Leak from external containment	1.5E-02	8.86E-03	8.038E-03	5.974E-03	4.743E-03	3.430E-03
－indirect leak (between internal containment and annular space)	1.336E-02	7.464E-03	6.670E-03	4.710E-03	3.579E-03	2.397E-03
－direct leak (between internal containment and environment)	1.635E-03	1.400E-03	1.368E-03	1.264E-03	1.165E-03	1.033E-03
Leak from external containment(between annular space and environment) －leak rate of 1%	1.183E-01	9.324E-02	8.987E-02	8.155E-02	7.674E-02	7.172E-02

－ Volume of the internal containment : 70,437m^3
－ Volume of the inter-containment annular space : 17,825m^3
－ Total volume retained by the external containment : 109,690m^3

tainment to inter-contain-ment) · (leak fraction from inter-containment to the environment), DF, is the decontamination factor of the filltration system for radionuclide i which is 1.0 for fission gases and 0.1 for radioiodines, and $R(t_a, t_b)$ is duration of release(h).

The total radiocativities released into the atmosphere are, then, the sum of the directly and indirectly released radioactivities.

$$QT_i(t_a,t_b) = QD_i(t_a,t_b) + QI_i(t_a,t_b) \cdots\cdots\cdots 3)$$

The time dependent leak rates for a 1300 MWe P'4 type containment.
are tabulated in table 3 and the total redioactivities released into the atmosphere are given in Table 4.

II.3 Atmospheric Dispersion

The "Les Abaques Globales" moel with 95% probaility of occurrence[8,9] is usually applied to obtain the atmospheric dispersion factor (χ/Q) in France for the distance greater than 100m from the release point[1,3,4].

For the control rooms which are not directly affected by the LOCA, and distance more than 100m from the LOCA reactor, the "Les Abaques globales" model are applied to calculate χ/Q at the point of external air-intake into these control rooms.

On the other hand, for the control room directly affected by the LOCA, since the distance from the stack to the external air-intake is generally less than 100m[3], the "Les Abaques Globales" model are not applicable. For this reason, the χ/Q value was calculated by the following semi empirical equation suggested in the Preliminary Safety Analysis Report of South Texas Project[1,3,4], which had been also app-

Table 4. Total radioactivities released into the atmosphere from containment building after a LOCA (Ci)-P'4 bype PWR.

Nuclide	1h	2h	6h	12h	24h	120h	240h	480h	720h
Kr 83m	17.10	27.10	44.44	48.64	49.11	49.12	49.12	49.12	49.12
Kr 85m	42.86	73.71	161.11	218.27	248.45	255.84	255.84	255.84	255.84
Kr 85	15.94	29.33	84.37	160.85	307.42	1855.05	3981/44	10157.20	16321.93
Kr 87	69.48	103.30	145.79	150.78	150.78	150.78	150.78	150.78	150.78
Kr 88	109.68	181.62	347.00	417.08	435.72	437.05	437.05	437.05	437.05
Sub total	255.06	414.96	782.70	995.43	1191.48	2747.84	4874.22	11049.98	17214.71
Xe 131m	1.09	2.00	5.74	10.86	20.46	109.70	204.30	384.15	484.77
Xe 133m	9.25	16.93	47.45	87.22	155.40	534.40	667.79	717.47	719.69
Xe 133	327.37	600.92	1709.69	3209.17	5943.61	27711.07	44416.28	63590.34	68764.25
Xe 135m	21.12	22.35	22.45	22.45	22.45	22.45	22.45	22.45	22.45
Xe 135	77.20	137.34	342.63	539.10	734.61	909.44	909.58	909.58	909.58
Xe 138	88.61	92.48	92.70	92.70	92.70	92.70	92.70	92.70	92.70
Sub total	524.64	872.02	2220.66	3961.50	6968.89	29379.76	46313.09	65716.62	70993.37
I 131	22.88	42.36	118.23	221.53	406.68	1552.68	2542.58	3685.59	4168.37
I 132	28.68	46.77	81.97	93.47	95.46	95.51	95/51	95.51	95.51
I 133	47.09	86.03	227.00	392.85	622.09	1044.53	1063.02	1063.41	1063.41
I 134	36.40	50.56	61.67	62.12	62.13	62.13	62.13	62.13	62.13
I 135	42.74	75.62	175.74	258.68	321.44	344.74	344.75	344.75	344.75
Sub total	177.79	301.35	664.60	1028.65	1507.80	3099.59	4107.98	5251.37	5734.16
Total	957.49	1588.33	3667.96	5985.58	9668.17	35227.18	55295.30	82017.87	93942.19

lied to the 900 MWe CP1 type PWR[1] in France.

$$\chi/Q = \frac{P \cdot K}{U \cdot A} \ (s/m^3) \cdots\cdots 4)$$

where, $K = 10/(S/D)^2$ and $0.5 < S/D < 3.0$[10,11], P is the probablity of wind from the reactor building toward the external air-intake of the reactor control room, S is the distance from the stack to the external air-intake of the reactor control room concerned(m), D is the diameter of the reactor building (m), U is the wind speed (m/s), and A is the cross-sectional area of the reactor building perpendicular to the wind direction(m^2).

The atmospheric dispersion factors calculated by Eq. (4) and by the "Les Abaques Globales" model for each control room are tabulated in Table 5.

II.4 Exposure Dose Assessment

1. Reactor Control Room

The reactor control room of a French 1300 MWe P'4 type PWR is located inside the auxilary building at a level of 15 m above the ground.

A system of ventilation and filtration supplies the filtered outdoor air of 3000 m^3/h to the reactor control room, which corresponds about 1 air change per

Table 5. Atmospheric dispersion factors(χ/Q).

Location	$\chi/Q(s/m^3)$	Duration	Remarks
· CR-1 (S=83 m)	9.80E-04	t < 1 d	evolutive, 1)
	3.27E-04	1 d < t < 10 d	
	1.64E-04	t > 10 d	
· CR-2 (S=125 m)	3.80E-04	constant	non-evolutive, 2)
· CR-3 (S=280 m)	1.25E-04	ditto	ditto
· CR-4 (S=430 m)	7.0E-05	ditto	ditto

1). from Eq. (4)
P : 1.0 for t<10 d and 0.5 otherwise,
U : 1.0m/s for t<24 h and 3.0 m/s otherwise,
D : 52m,
2) from Le Quinio's 95% "Les Abaques Globales" model.

hour.

The minimum iodine removal efficiency of the charcoal bed type filter, which is in series of the particulate filter is conservatively assumed to be 90% for all types of iodines[3]. The filtration system assumed to be in operation at the time of LOCA breakout and be not interrupted during the accident period.

2. Principal Exposure Modes

The principal exposure modes to the reactor operator inside the reactor control room are as follows in decreasing order of importance :

1) Internal thyroid exposure : inhalation of iodines
2) External whole body exposure(I) : immersion in the contaminated air
3) External whole body exposure(II) : direct radiation from the fission products retained in the reactor containment building.

Among these three modes of exposures, mode 3) is assumed to be negligible compared to the other modes, because of the distance and the shielding effect of the thick concrete between the reactor building and and the reactor control room[1,3].

3. Exposure in the Reactor Control Room

By assuming that a control room has the geometric form of semi-finite sphere in which contaminated air is uniformly distributed, the gamma absorbed dose rate to a reactor operator in the center of this sphere can be obtained by the following gamma fluence equation[10,11,12] :

$$ {}_{\gamma}\dot{D}_{R/2} = 0.254\chi \cdot E_{\gamma} \cdot \mu_a \{1-(1+k\mu_a R)\exp(-\mu R)\} \cdots\cdots 5) $$

where, $\gamma\dot{D}_{R/2}$ is gamma absorbed dose rate at the center of a semi-finite sphere of radius R(rad/s), χ is the fission gas concentration in air (Ci/cm^3), E_γ is the average gamma energy of radionuclide (MeV), f_γ is the fraction of gamma emission, μ is the linear energy attenuation coefficient in air(m^{-1}), μ_a is the linear energy absorption coefficient in air (m^{-1}) and R is the radius of the semi-sphere of which volume is equivalent to that of the reactor control room.

The gamma absorbed dose rate in a semi-infinite spherical cloud is calculated by the following equation [13].

$${}_{\gamma}\dot{D}_{\infty/2}=0.25\chi \cdot E_{\gamma} \cdot f_{\gamma}(rad/s) \cdots\cdots 6)$$

where, ${}_{\gamma}\dot{D}_{\infty}/2$: absorbed gamma dose rate in the a semi-infinite spherical air(rad/s)

The volume factor(VF) which tells the extent of dose reduction due to the finite volume of the reactor control room is the ratio of ${}_{\gamma}\dot{D}_{R/2}$ to ${}_{\gamma}\dot{D}_{\infty/2}$.

$$VF={}_{\gamma}\dot{D}_{R/2}/{}_{\gamma}\dot{D}_{\infty/2}$$
$$=1.015\{1-(1+k\mu_{a}R)\exp(-\mu R)\} \cdots\cdots 7)$$

Thus the semi-finite dose conversion factor within a limited space is obtained from the correction of $DCF_{\infty/2}$ by the volume factor of each radionuclide.

Table 6 shows the semi-infinite dose conversion factors($DCF_{\infty/2}$) for external whole body exposure and internal thyroid exposure[14].

Table 6. Semi-infinite dose conversion factors(DCF) in air (rem · m^3 · Ci^{-1} · s^{-1}).

Nuclide	External Exposure(*) (whole body)	Iodine Inhalation(**) (thyroid) t < 24h	t > 24h
Kr 83m	2.10E-03		
Kr 85m	4.00E-02		
Kr 85	5.57E-04		
Kr 87	0.174		
Kr 88	0.493		
Xe 131m	4.82E-03		
Xe 133m	5.45E-02		
Xe 133	1.11E-02		
Xe 135m	0.108		
Xe 135	6.22E-02		
Xe 138	0.264		
I 131	9.51E-02	370	290
I 132	0.597	2.1	1.6
I 133	0.154	60	47
I 134	0.658	0.37	0.29
I 135	0.410	14	11

* Report CEA-R-4284 [1]
** RSS P4 Chap. III.

The semi-finite absorbed dose rate(${}_{\gamma}D_{R/2}$)and the volume factor in this study are given in Table 7 with those proposed by EDF[3] for the purpose of comparison.

4. Time-Integrated Exposure Dose

The time-integrated external whole body and thyroid doses for a reactor operator in the control room can be calculated by

$$H(t)=\sum_{i} Q_i(t) \cdot \chi/Q(t) \cdot DCF_{i\infty} \cdot VF_i \cdot DF_i(rem) \cdots\cdots 8)$$

where, H(t) is the time-integrated exposure dose to

Table 7. Gamma dose rate in the semi-infinite and semi-finite air (rad/s) and volume reduction factors

Nuclide	$\gamma \dot{D}_{\infty}$	$\gamma \dot{D}_{11.4}$	VF[1)]	VF-EDF[3][2)]	1)/2)
Kr 83m	8.03E-03	1.53E-03	0.1905	0.193	0.987
Kr 85m	3.96E-02	1.01E-03	0.0482	0.047	1.025
Kr 85	5.58E-03	2.62E-05	0.0470	0.042	1.119
Kr 87	0.174	7.87E-03	0.0451	0.036	1.253
Kr 88	0.493	1.76E-02	0.0356	0.314	0.113
Xe 131m	4.82E-03	3.78E-04	0.782	0.178	0.439
Xe 133m	5.45E-02	2.67E-04	0.0490	0.110	0.445
Xe 133	1.11E-02	1.10E-03	0.0987	0.095	1.039
Xe 135m	0.108	5.08E-03	0.0470	0.050	0.904
Xe 135	6.22E-02	2.90E-03	0.0466	0.052	0.896
Xe 138	0.264	1.01E-02	0.0381	0.035	1.088
I 131	9.75E-02	4.63E-03	0.0475	0.043	1.104
I 132	0.597	1.96E-02	0.0328	0.041	0.800
I 133	0.154	7.02E-03	0.0455	0.043	1.058
I 134	0.658	2.12E-02	0.0322	0.041	0.785
I 135	0.410	1.52E-02	0.0370	0.037	1.000

whole body or thyroid at time t after a LOCA(rem), $Q_i(t)$ is the released radioactivity into the atmosphere at time t after the LOCA (Ci), $Q_i(t) \cdot \chi/Q(t)$ is the time-integrated concentration of fission products which enter the control room at time t,(Ci · s/m^3) and $DCF_{i\infty}$ is the semi-infinite dose conversion factor for radionuclide i for whole body or thyroid exposure (rem · m^3/Ci · s).

III. Results

A computer code COREX (COntrol Room EXposure) was developed to calculate the dose to a reactor operator due to submersion in contaminated air and inhalation of radioiodines within a control room after a LOCA.

The time-integrated radioactivities released into the atmosphere, the volume factors, and the resulting semi-finite dose conversion factors can be calculated by use of COREX.

In Table 8, the time-integrated doses to the whole body and the thyroid of a reactor operator are listed for various time intervals up to 30 days following a LOCA and these doses are shown in Figure. 3 and 4 with the data proposed by EDF[3].

IV. Conclusions

1. The time-integrated dose to the whole body of a reactor operator up to 30 days following a LOCA is less than 0.4 mSv(0.04 rem).

2. The time-integrated thyroid dose to a reactor operator due to inhalation of radioiodines is 480 mSv (48 rem) after 30 days following a LOCA, which is far below the dose equivqlent limit for reactor accident, 3 Sv(300 rem).

3. However, in viewpoint of the control habitability during the emergency operational period, the thy-

Table 8. Time-integrated whole body and thyroid doses in different reactor control rooms for a P'4 Type PWR after a LOCA.

Local	Organ	1h	2h	6h	12h	24h	120h	240h	480h	720h
CR-1	W.B.	0.0043	0.0066	0.0121	0.0161	0.0205	0.0241	0.0304	0.0304	0.0305
(83m)	Thy.	1.1170	2.1600	5.8800	1.0700	18.900	18.900	28.200	28.200	28.200
CR-2	W.B.	0.0068	0.0025	0.0047	0.0062	0.0079	0.0179	0.0252	0.0336	0.0359
(125m)	Thy.	0.4550	0.8630	2.2800	4.1600	7.3200	19.200	30.100	42.700	48.000
CR-3	W.B.	0.0005	0.0008	0.0015	0.0020	0.0020	0.0059	0.0083	0.0110	0.0118
(280m)	Thy.	0.1500	0.2750	0.7500	1.3700	2.4101	6.3000	9.9900	14.200	15.800
CR-4	W.B.	0.0003	0.0004	0.0008	0.0011	0.0014	0.0033	0.0046	0.0618	0.0661
(430m)	Thy.	0.0830	0.1540	0.4200	0.7600	1.3500	3.5300	5.5400	7.8600	8.8400

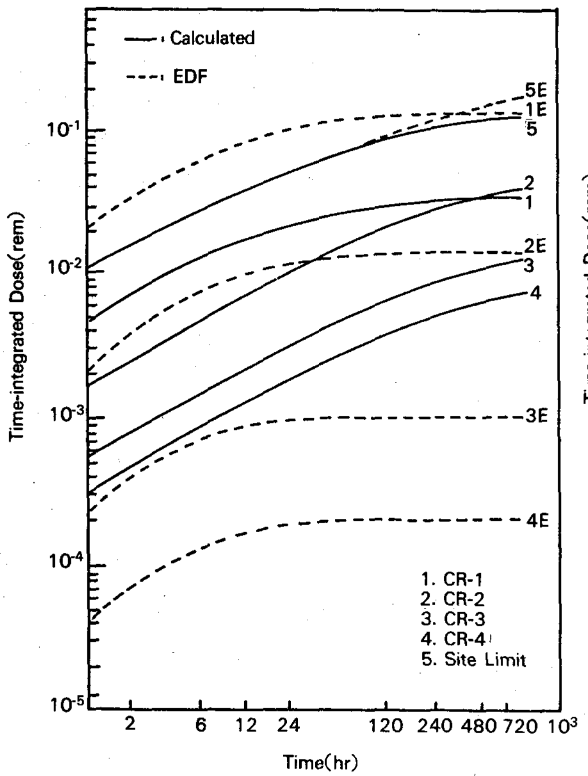

Fig. 3. Time-integrated whole body dose after a LOCA in the 1300 MWe P'4 type PWR.

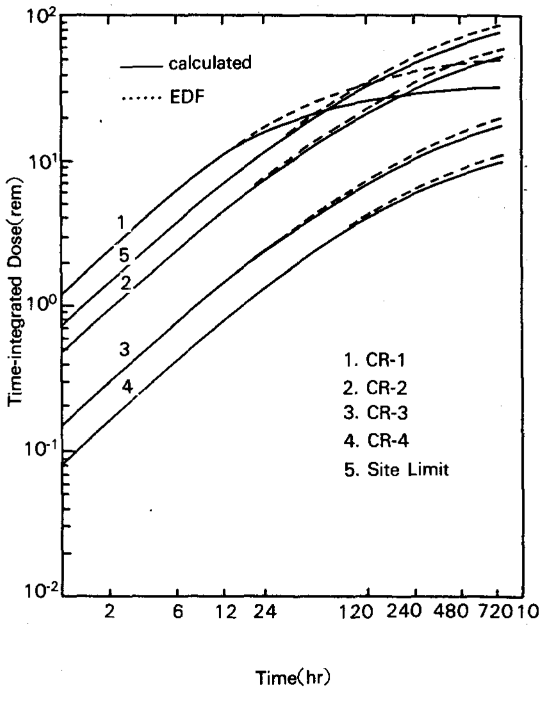

Fig. 4. Time-integrated thyroid dose after a LOCA in the 1300 MWe P'4 type PWR.

roid dose to an operator could be substantially reduced by more than 50% if we apply at least 2 work-shift condition a day and improve the radioiodine removal efficiency of the control room filtration system from 90% to 95%.

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冷却材 喪失事故時 1300 MWe 級 PWR 原電 主制御室의 線量評價

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要 約

프랑스의 1300 MWe 級 標準 P'4형 PWR 原電의 一次冷却材喪失事故(LOCA)時 原電 主制御室內 運轉員에 대한 放射線 被瀑線量을 계산하여 主制御室의 滯留安全性을 評價하였다. 本 評價에서 使用된 諸假定은 프랑스의 標準安全性分析報告書에 따랐다. 本 評價를 위하여 LOCA 事故時 原子爐建物外로 放出되는 放射核種의 放射能, 主制御室에서의 體積因子 및 制御室內 運轉員의 全身 및 甲狀腺 被瀑線量을 事故發生後 30일까지 電算할 수 있는 간단한 電算프로그램, COREX를 開發하였다.

本 硏究에서 얻어진 計算結果는 대체적으로 프랑스의 EDF(불란서 電力株式會社) 에서 提案한 結果와 대체적으로 잘 一致하였으나, 全身外部被瀑線量의 값은 一部 體積因子 값의 차이로 因하여 일부 偏差를 보였다.