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ESTIMATION OF OFF-SITE DOSE AND RELEASE CONCENTRATION OF RADIOACTIVE LIQUID EFFLUENTS FROM RADWASTE TREATMENT SYSTEM IN KORI 3&4

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Abstract - The designed release rate of liquid effluents from radwaste treatment system should be calculated and evaluated during normal operation, including anticipated operational occurrence and be assured that the release concentration and off-site dose at unrestricted area do not exceed the limits of regulation. The expected annual release rate and off-site dose for the currently operating nuclear power plants in Korea had been calculated and evaluated using PWR-GALE and LADTAP-II which was based on USNRC Regulatory Guide 1.109. Recently, the MOST Notice 2001-2 related to release concentration and off-site dose at unrestricted area was revised to reflect the concept of ICRP-60. It is necessary for KORI 3&4 to re-calculate the release concentration and off-site dose and to compare these results with the limits of regulation. As the results of assessment, we confirmed that the release concentrations were less than its limits of MOST Notice 2001-2 and the off-site dose at unrestricted area using K-DOSE60 was 3.61E-03 mSv/yr to the age of five for the effective dose, and 4.10E-2 mSv/yr to thyroid of the age of five for the organ equivalent dose. We also confirmed the off-site dose was within the limits of MOST Notice 2001-2. Therefore, the release concentration and off-site dose re-evaluated at unrestricted area in KORI 3&4 were well below the regulation limits of MOST Notice 2001-2.

INTRODUCTION

The sources of liquid radwaste in nuclear power plant are fission products and reactor coolant containing activated products of system structures. The inputs to liquid radwastes are effluents from reactor coolant purification process, leakages from pumps, valves and equipments, and laundry and hot shower. Fluids entering the waste disposal system are segregated and collected according to their activities and chemical form. The liquid radwastes generated from purification process of reactor coolant contain a lot of boron as well as radioactive materials in the form of particles or ions and are reused or discharged through filter and boron evaporator. Also, the liquid radwastes collected in sump are pumped into holdup tank and then treated by filter, ion exchanger or evaporator to discharge into the environment. The designed release rate of liquid effluents in the liquid radwaste treatment system should be calculated and evaluated during normal operation, including anticipated operational occurrence and be assured that the release concentration and off-site dose at unrestricted area do not exceed the limits of regulation. Generally, in order to meet these requirements, the licensee should evaluate release concentration and off-site dose of effluents discharged from nuclear power plant using the expected source term and PWR-GALE (NUREG- 0017, April 1976) program. The limits of regulation for release concentration and off-site dose at unrestricted area are described in MOST Notice. The expected annual release rate and off-site dose for the currently operating nuclear power plants in Korea had been calculated and evaluated using PWR-GALE and LADTAP-II which was based on USNRC Regulatory Guide 1.109. PWR-GALE (NUREG-0017, Rev.1) program was revised in 1985, but this revised program was not applied for KORI 1,2,3&4 and YGN1&2 cases. Recently, the MOST Notice 2001-2 related to release concentration and off-site dose at unrestricted area was revised to reflect the concept of ICRP-60. Therefore, the K-DOSE60 program considering the concept of ICRP-60 was developed by KEPRI(Korea Electric Power Research Institute). It is necessary for KORI 3&4 to re-calculate the release concentration and off-site dose using the revised PWR-GALE and K-DOSE60 program and to compare these results with the limits of MOST Notice 2001-1.

CALCUALTION OF EXPECTED RELEASE RATE

The expected annual liquid activity release of KORI 3&4 was calculated using the expected source term and PWR-GALE(NUREG-0017, Rev.1). Input parameters of PWR-GALE program were refered from FSAR(Final Safety Analysis Report) of KORI 3&4 and some of them were also refered from NUREG-0017, Rev.1. Table 1 shows the input parameters of PWR-GALE and Table 2 indicates the annual releases rate of the liquid waste discharged from KORI 3&4. For PWR-GALE coding, total six liquid radwaste inlet streams were considered as follows:

- ㅇ Shim Bleed
- ㅇ Equipment Drain Waste
- ㅇ Clean Waste
- ㅇ Dirty Waste
- ㅇ Blowdown Waste
- ㅇ Regenerant Waste

Table 1. Input parameters used in PWR-GALE

CARD	1	NAME	NAME OF REACTOR KRN34	PWR
CARD	2	POWTH	THERMAL POWER LEVEL (MEGAWATTS)	2900.
CARD	3	PCVOL	MASS OF PRIMARY COOLANT (THOUSAND LBS)	397.0
CARD	4	LETDWN	PRIMARY SYSTEM LETDOWN RATE (GPM)	75.0
CARD	5	CBFLR	LETDOWN CATION DEMINERALIZER FLOW (GPM)	7.5
CARD	6	NOGEN	NUMBER OF STEAM GENERATORS	3.
CARD	7	TOSTFL	TOTAL STEAM FLOW (MILLON LBS/HR)	12.3
CARD	8	WLI	MASS OF LIQUID IN EACH STEAM GENERATOR (THOUND LBS)	110.
CARD	9	BLWDWN	BLOWDOWN-THOUSLB/HR 123.00 BLOWDOWN TREATMENT-INPUT	0
CARD	10	REGENT	CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)	2.4
CARD	11	FFCDM	CONDENSATE DEMINERALIZER FLOW FRACTION	0.34
CARD	12		SHIM BLEED RATE 1073. GPD AT 1.0 PCA	
CARD	13		DFI= 1.0E05DFCS= 2.0E03 DFO= 1.0E04	
CARD	14		COLLECTION 50.8 DAYS PROCESS 3.100 DAYS FRACT DISCH	0.10
CARD	15		EQUIPMENT DRAINS INPUT 250. GPD AT 0.70 PCA	
CARD	16		DFI= 1.0E05DFCS= 2.0E03 DFO= 1.0E04	
CARD	17		COLLECTION 50.8 DAYS PROCESS 3.100 DAYS FRACT DISCH	0.10
CARD	18		CLEAN WASTE INPUT 675. GPD AT 0.001PCA	
CARD	19		DFI= 1.0E05DFCS= 2.0E04 DFO= 1.0E06	
CARD	20		COLLECTION 17.8 DAYS PROCESS 0.278 DAYS FRACT DISCH	1.00
CARD	21		DIRTY WASTE INPUT 1710. GPD AT 0.076PCA	
CARD	22		DFI= 1.0E05DFCS= 2.0E04 DFO= 1.0E06	
CARD	23		COLLECTION 14.0 DAYS PROCESS 0.556 DAYS FRACT DISCH	1.00
CARD	24		BLOWNDOWN FRACTION PROCESSED	1.0
CARD	25		DFI= 1.0E03DFCS= 1.0E02 DFO= 1.0E03	
CARD	26		COLLECTION 0.0 DAYS PROCESS 0.0 DAYS FRACT DISCH	0.1
CARD	27		REGENERANT FLOW RATE (GPD)	3400.
CARD	28		DFI= 1.0E00DFCS= 1.0E00 DFO= 1.0E00	
CARD	29		*COLLECTION 0.00 DAYS PROCESS 0.000 DAYS FRACT DISCH*	*1.0*
CARD	30		IS THERE CONTINUOUS STRIPPING OF FULL LETDOWN FLOW?	1
CARD	31	TAU1	HOLDUP TIME FOR XENON (DAYS)	77.2
CARD	32	TAU2	HOLDUP TIME FOR KRYPTON (DAYS)	5.98
CARD	33	TAU3	FILL TIME OF DECAY TANKS FOR THE GAS STRIPPER (DAYS)	
CARD	34		GAS WASTE SYSTEM HEPA?99.	
CARD	35		FUEL BLDG CHARCOAL? 00.HEPA? 00.	
CARD	36		AUXILIARY BLDG CHARCOAL? 90.HEPA? 99.	
CARD	37	CONVOL	CONTAINMENT VOLUME (MILLION FT3)	2.08
CARD	38		CNTMT ATM. CLEAN-UP CHARCOAL? 00.HEPA? 00. RATE(1000CFM)	0.0
CARD	39		CNTMT-HIGH VOL PURGECHARCOAL? 0.0HEPA? 00.	
CARD	40		CNTMT LOW VOL PURGE CHARCOAL? 90.HEPA? 99. RATE (CFM)	4000.
CARD	41	FVN	FRACTION IODINE RELEASED FROM BLOWDOWN TANK VENT	0.01
CARD	42	FEJ	PERCENT OF IODINE REMOVED FROM AIR EJECTOR RELEASE	0.0075
CARD	43	PFLAUN	DETERGENT WASTE PF	1.0

The inputs related to the liquid radwaste streams except for blowdown and regenerant waste are calculated as follows.

1. SHIM BLEED

The shim bleed is a letdown flow entering boron recycle system to collect and process reactor coolant grade wastes prior to volume control tank in chemical and volume control system. The input parameters related to shim bleed are as follows.

- Flow rate and PCA(Primary Coolant Activity) : 1073 gallon/day, 1.0 PCA
- Collection Time
  Tc = 0.8 X Holdup Tank Vol./Shim Bleed Flow+(Equipment Drain+Reactor Drain)rate [for Redundant]
  = 0.8 X 84,000 gal/(1073+200+50) gpd = 50.8days
- Processing Time
  Tp = 0.8 X Holdup Tank Vol./Equipment Flow Capacity [for Redundant]
  = 0.8 X 84,000 gal/(15gpm X 1440mpd)
  = 3.1days
- Decontamination Factor (DF)
  Iodine = 10^5, Cs, Rb = $2x10^3$, Others = 10^4

2. EQUIPMENT WASTE

The equipment waste represents an equipment drain flow with reactor grade wastes. For KORI 3&4, this means flow entering boron recycle system from reactor coolant drain tank and equipment drain tank. These flows combined with the shim bleed and the treatment process is the same as shim bleed. This flow rates were obtained from Korea standard nuclear power (KSNP) plant because we couldn't obtain them from KORI 3&4 FSAR. The input parameters related to the equipment waste are as follows.

- Flow rate and PCA

Input source	Flow rate (gpd)	PCA
Reactor Coolant Drain Tank	200	0.62
Equipment Drain Tank	50	1.0
Total	250	0.7

- Collection Time(Tc) and Processing Time(Tp) : same as shim bleed
- Decontamination Factor (DF) : same as shim bleed

Table 2. Results of expected release rate

Nuclides	Primary Coolant (μCi/mℓ)	Release Rate (Ci/yr)
Na-24	4.48E-02	9.20E-03
P-32	0.00E+00(*)	1.80E-04
Cr-51	2.62E-03	6.40E-03
Mn-54	1.34E-03	4.70E-03
Fe-55	1.01E-03	7.90E-03
Fe-59	2.53E-04	2.40E-03
Co-58	3.87E-03	1.10E-02
Co-60	4.45E-04	1.40E-02
Ni-63	0.00E+00	1.70E-03
Zn-65	4.29E-04	2.80E-04
W-187	2.30E-03	6.80E-04
Np-239	1.94E-03	8.90E-04
Br-84	1.85E-02	1.40E-04
Rb-88	2.23E-01	1.00E-04
Sr-89	1.18E-04	1.60E-04
Sr-90	1.01E-05	2.00E-05
Sr-91	9.49E-04	1.30E-04
Y-91m	5.28E-04	7.80E-05
Y-91	4.38E-06	9.00E-05
Y-93	4.13E-03	5.70E-04
Zr-95	3.28E-04	1.30E-03
Nb-95	2.36E-04	2.10E-03
Mo-99	5.59E-03	2.80E-03
Tc-99m	4.84E-03	2.40E-03
Ru-103	6.32E-03	4.40E-03
Rh-103m	0.00E+00	4.10E-03
Ru-106	7.56E-02	5.90E-02
Rh-106	0.00E+00	5.00E-02
Ag-110m	1.09E-03	1.90E-03
Te-129m	1.60E-04	1.00E-04
Te-129	2.73E-02	2.70E-04
Te-131m	1.36E-03	4.70E-04
Te-131	8.96E-03	9.60E-05
I-131	3.86E-02	8.10E-02
Te-132	1.48E-03	7.40E-04
I-132	2.31E-01	1.90E-02
I-133	1.31E-01	1.20E-01
I-134	3.90E-01	6.80E-03
Cs-134	5.97E-03	1.40E-02
I-135	2.66E-01	8.30E-02
Cs-136	7.44E-04	7.00E-04
Cs-137	7.90E-03	2.00E-02
Ba-137m	0.00E+00	3.90E-03
Ba-140	1.10E-02	7.60E-03
La-140	2.23E-02	1.10E-02
Ce-141	1.26E-04	3.10E-04
Ce-143	2.53E-03	9.00E-04
Pr-143	0.00E+00	6.30E-05
Ce-144	3.28E-03	6.10E-03
Pr-144	0.00E+00	2.20E-03
H-3	3.5E+00	3.50E+02

*Note *) 0.00 : indicates that the value is less than 1.00E-06.*

3. CLEAN WASTE

This waste means normally tritiated, nonaerated, low-conductivity liquids consisting primarily of liquid waste collected from equipment leaks and drains, and certain valve and pump seal leakoffs according to definition of NUREG-0017, Rev.1. But, it is hard to distinguish the clean wastes from radwastes entering liquid radwaste treatment system in KORI 3&4. This plant collects liquid radwastes in the high- and low-TDS(Total Dissolved Dolids) holdup tanks based on concentration and radioactivity in dissolved materials. Conventionally, we consider that the liquid radwaste entering the low-TDS holdup tank is clean waste and the high-TDS holdup tank is dirty waste. According to the FSAR of KORI 3&4, the liquid sources entering the low-TDS holdup tank are radwaste solidification system, radwaste tunnel normal sump, radwaste bldg. resin sluice pump, LRS adsorption bed, auxiliary steam condensate recover tank, LRS monitor tank and LRS evaporator off-spec. distillate. In order to obtain the flow rate and PCA of clean waste, we should know the flow rate and PCA from above liquid source entering the low-TDS holdup tank. Actually, it was difficult to obtain them from KORI 3&4 and these liquid sources are not mentioned in Table 1-3 of NUREG-0017, Rev.1. Thus, we utilized the flow rate of miscellaneous waste mentioned in ANSI/ANS-55.6(1993) in order to obtain the input flow and PCA of the clean waste like waste inputs which were used to calculate the flow rate and PCA of the clean wasts in KSNP.

- Flow rate and PCA : 675 gpd, 0.001 PCA
- Collection Time
  Tc = 0.4 X 30,000 gal/675gpd = 17.8days
- Processing Time
  Tp = 0.4 X 30,000 gal/30gpm X 1440mpd
  = 0.278days
- Decontamination Factor (DF)
  Iodine = 10^5, Cs, Rb = $2x10^3$, Others = 10^6

4. DIRTY WASTE

This waste means normally nontritiated, aerated, high-conductivity, non primary-coolant quality liquids collected from building sumps and floor, and sample station drains. We used the values in Table 1-3 of NUREG-0017, Rev.1 as input sources entering the high-TDS holdup tank.

- Flow rate and PCA

Input source	Flow rate (gpd)	PCA
Containment Sump		
- Primary Coolant Pump Seal Leakage	20	0.1
- Primary Coolant Leakage	10	1.67
- Primary Coolant Equipment Leakage	500	0.001
Aux. Bldg. Sump		
- Primary Coolant System Equipment Drains	80	1.0
- Primary Coolant Sampling System Drains	200	0.05
- Aux. Bldg. Floor Drains	200	0.1
Fuel Bldg. Sump		
- Fuel Bldg. Normal Sump	700	0.001
Total	1710	0.076*

- Collection Time
  Tc = 0.8 X 30,000 gal/1710gpd
  = 14.0 days [for Redundant]
- Processing Time
  Tp = 0.8 X 30,000 gal/30gpm X 1440mpd
  = 0.556days [for Redundant]
- Decontamination Factor (DF) : same as clean waste

CONCENTRATION AND FRACTION OF EFFLUENTS

The concentrations of radioactive liquid effluents at unrestricted area were calculated as follows.

$$C_{unres(i)} = \frac{R(i) \cdot MF(i)}{F_{dil} \cdot CF \cdot DF} \qquad (1)$$

where,

$C_{unres}(i)$: Concentration at unrestricted area of nuclide i (μCi/mℓ)
R(i) : Total annual release rate of the nuclide i (Ci/yr)
MF(i) : Multiplication Factor for the nuclide I

$$MF(i) = \frac{RCS(i)_{1\%}}{RCS(i)_{PWR-GALE}} \qquad (2)$$

$RCS(i)_{1\%}$: Concentration of nuclide i in primary coolant based on 1 % fuel failure
$RCS(i)_{PWR-GALE}$: Concentration of nuclide i

in primary coolant calculated from PWR-GALE

F_{dil} : Dilution flow rate (1,404 ft^3/sec : KORI 3&4 FSAR)

CF : Conversion factor(8.93E+05cm^3 · Ci · sec/ft^3 · μCi · yr)

DF : Dilution factor at unrestricted area (700 meter for KORI 3&4)

The source term within reactor coolant system was based on the assumption of the design basis reactor coolant system with equilibrium concentration resulting from fission product leakage into the reactor coolant system based on 1% fuel failure in accordance with the Standard Review Plan Section 11.3. For this reason, multiplication factor is applied for this calculation. The fraction of the effluent concentration (FEC) for each nuclide was calculated as follows.

$$FEC_i = \frac{C_i}{EC_i} \tag{3}$$

where,

Table 3. Concentrations of nuclide in primary coolant system based on 1% fuel failure[a)]

Nuclides	Activity (μCi/mℓ)	Nuclides	Activity (μCi/mℓ)	Nuclides	Activity (μCi/mℓ)
H-3	3.5	I 135	2.1	Kr 85(*)	7.7
Br-84	4.9E-2	Te 129m	1.5E-2	Kr 85m(*)	2.0
Rb-88	4.7	Te 129	1.6E-2	Kr 87(*)	1.3
Rb-89	2.2E-1	Te 131m	2.2E-2	Kr 88(*)	3.7
Sr-89	3.5E-3	Te 131	1.2E-2	Kr 89	0.11
Sr-90	9.9E-5	Te 132	0.24	Xe 131m(*)	2.1
Sr-91	5.7E-3	Te 134	3.1E-2	Xe 133(*)	2.6E+2
Sr-92	1.3E-3	Cs 134	1.8	Xe 133m(*)	1.7E+1
Y-90	2.6E-5	Cs 136	3.0	Xe 135(*)	7.2
Y-91m	3.1E-3	Cs 137	1.2	Xe 135m(*)	0.47
Y-91	4.6E-4	Cs 138	1.0	Xe 137	0.18
Y-92	1.0E-3	Ba 137m	1.1	Xe 138(*)	0.66
Y-93	3.4E-4	Ba 140	3.4E-3	Cr 51	5.5E-3
Zr-95	5.3E-4	La 140	1.0E-3	Mn 54	4.0E-4
Nb-95	5.3E-4	Ce 141	5.1E-4	Mn 56	2.2E-2
Mo-99	6.3E-1	Ce 143	4.3E-4	Fe 55	2.3E-3
I-131	2.3	Ce 144	3.1E-4	Fe 59	5.8E-4
I-132	2.8	Pr 143	5.2E-4	Co 58	1.5E-2
I-133	3.7	Pr 144	3.1E-4	Co 60	1.9E-3
I-134	0.59	Kr 83m(*)	4.6E-1		

Note) a refer to KORI FSAR
** indicate noble gas*

FEC_i : Fraction of EC for nuclide i
EC_i : EC of the nuclide i (MOST Notice 2001-2 Table 3 Column 8)

The concentrations of nuclide in the primary coolant system based on 1% fuel failure are shown in Table 3 and the released concentration compared with MOST Notice 2001-2 are shown in Table 4.

Table 4. Release concentration and comparison with MOST Notice 2001-2

Nuclide	MF	Concentration (Bq/m^3)	MOST Notice 2001-2(Bq/m^3)	Fraction(FEC)
Na-24	1.00E+00	1.36E-01	2.00E+06	6.81E-08
P-32	1.00E+00	2.67E-03	3.00E+05	8.89E-09
Cr-51	1.00E+00	9.48E-02	2.00E+07	4.74E-09
Mn-54	1.00E+00	6.96E-02	1.00E+06	6.96E-08
Fe-55	1.00E+00	1.17E-01	2.00E+06	5.85E-08
Fe-59	1.00E+00	3.55E-02	4.00E+05	8.89E-08
Co-58	1.00E+00	1.48E-01	9.00E+05	1.65E-07
Co-60	1.00E+00	2.07E-01	2.00E+05	1.04E-06
Ni-63	1.00E+00	2.52E-02	5.00E+06	5.04E-09
Zn-65	1.00E+00	4.15E-03	2.00E+05	2.07E-08
W-187	1.00E+00	1.01E-02	3.00E+05	3.36E-08
Np-239	1.00E+00	1.32E-02	9.00E+05	1.46E-08
Br-84	2.65E+00	5.49E-03	8.00E+06	6.87E-10
Rb-88	2.11E+01	3.12E-02	8.00E+06	3.90E-09
Sr-89	2.97E+01	7.03E-02	3.00E+05	2.34E-07
Sr-90	9.80E+00	2.90E-03	2.00E+04	1.45E-07
Sr-91	6.01E+00	1.16E-02	9.00E+05	1.29E-08
Y-91m	5.87E+00	6.78E-03	6.00E+07	1.13E-10
Y-91	1.05E+02	1.40E-01	3.00E+05	4.67E-07
Y-93	8.23E-02	6.95E-04	6.00E+05	1.16E-09
Zr-95	1.62E+00	3.11E-02	8.00E+05	3.89E-08
Nb-95	2.25E+00	6.99E-02	1.00E+06	6.99E-08
Mo-99	1.13E+02	4.67E+00	6.00E+05	7.79E-06
Tc-99m	1.18E+02	4.19E+00	3.00E+07	1.40E-07
Ru-103	7.28E-02	4.74E-03	9.00E+05	5.27E-09
Rh-103m	4.60E+01	2.79E+00	2.00E+08	1.40E-08
Ru-106	1.46E-03	1.27E-03	1.00E+05	1.27E-08
Rh-106m	1.10E+01	8.15E+00	4.00E+06	2.04E-06
Ag-110m	1.01E+00	2.84E-02	2.00E+05	1.42E-07
Te-129m	9.38E+01	1.39E-01	2.00E+05	6.94E-07
Te-129	5.86E-01	2.34E-03	1.00E+07	2.34E-10
Te-131m	1.62E+01	1.13E-01	4.00E+05	2.82E-07
Te-131	1.34E+00	1.90E-03	8.00E+06	2.38E-10
I-131	5.96E+01	7.15E+01	3.00E+04	2.38E-03
Te-132	1.62E+02	1.78E+00	2.00E+05	8.89E-06
I-132	1.21E+01	3.41E+00	2.00E+06	1.71E-06
I-133	2.82E+01	5.02E+01	2.00E+05	2.51E-04
I-134	1.51E+00	1.52E-01	6.00E+06	2.54E-08
Cs-134	3.02E+02	6.25E+01	4.00E+04	1.56E-03
I-135	7.89E+00	9.71E+00	7.00E+05	1.39E-05
Cs-136	4.03E+03	4.18E+01	2.00E+05	2.09E-04
Cs-137	1.52E+02	4.50E+01	5.00E+04	9.00E-04
Ba-137m	1.10E+05	6.35E+03	3.00E+05	2.12E-02
Ba-140	3.09E-01	3.48E-02	3.00E+05	1.16E-07
La-140	4.48E-02	7.31E-03	3.00E+05	2.44E-08
Ce-141	4.05E+00	1.86E-02	1.00E+06	1.86E-08
Ce-143	1.70E-01	2.27E-03	6.00E+05	3.78E-09
Pr-143	5.20E+01	4.85E-02	6.00E+05	8.09E-08
Ce-144	9.45E-02	8.54E-03	1.00E+05	8.54E-08
Pr-144	3.10E+01	1.01E+00	1.00E+07	1.01E-07
H-3	1.00E+00	5.72E+03	2.00E+07	2.86E-04
total	5.63E+00	3.22E+04	4.10E+08	7.86E-05

EVLUATION OF OFF-SITE DOSE

A program, K-DOSE60, was utilized to evaluate radiological exposure due to the release of radioactive material from nuclear power plants during normal operation via liquid effluent pathways. This program was implemented for the radiological exposure models as described in USNRC Regulatory Guide 1.109, Rev.1 for radioactivity releases in liquid effluents and developed in KEPRI and reflected the concept of ICRP-60. It has been used for reactor licensing evaluations of YGN 5&6 and to estimate maximum individual and general population doses. The usage factors contained in K-DOSE60 utilized KORI ODCM(Off-site Dose Calculation Manual) data and the final report of environment assessment in KORI, 1989. Table 5 represents the major assumptions used in dose analysis and Tables 6 and 7 represent ocean activity time and intake and ocean activity time of maximum/average individual.

Table 5. Major Assumptions used in Dose Analysis

Items	Assumption
Dose Receptor	Adult, Teenager(age 15), Child(age 5) (No significant pathways to infant receptors)
Dose Pathway	- Aquatic foods (Fish, Mollusca, Crustacea, Algae) - Shoreline - External exposures (Swimming, Boating)
Bioaccumulation	Saltwater values from Regulatory Guide1.109 where applicable
DCF*	ICRP 67, 69, 72 and the ICRP database of Dose Coefficients:Works and Members of the Public 1999
Organs (MOST Notice 2001-2)	Bladder, Bone surface, Breast, Oesophagus, Large intestine, Stomach, Liver, Ovaries, Marrow(Red), Lungs, Skin, Thyroid, Other organ
Dilution Factor	2

DCF is Dose Conversion Factor

Table 6. Ocean Activity Time

Point	Shoreline (hr/yr)	Swimming (hr/yr)	Boating(hr/yr)
ILKWANG	1.46E+5	4.40E+5	2.06E+7
KIJANG	5.13E+5	1.54E+6	7.21E+7
HYOAM	1.02E+5	3.07E+5	1.44E+7
HAEUNDAE	5.47E+6	1.64E+7	7.70E+8
BUSAN	0	0	0
WOLRAE	1.58E+5	4.75E+5	2.22E+7

Table 7. Intake and Ocean Activity Time of Maximum/ Average Individual (hr/yr)

Age / Pathways	Child (5 year-old)		Teenage (15 year-old)		Adult	
	Maximum	Average	Maximum	Average	Maximum	Average
Fish(kg/yr)	52.9	60.5	82.8	118.4	79.3	113.4
Mollusca(kg/yr)	5.9	5.5	9.2	10.7	8.8	10.3
Crustacea(kg/yr)	5.9	5.5	9.2	10.7	8.8	10.3
Algae(kg/yr)	10.6	8.7	16.6	17	15.8	16.3
Shoreline(hr/yr)	14.0	9.5	67	47	12	8.3
Swimming(hr/yr)	300	142	240	106	60	21
Boating(hr/yr)	-	-	-	-	3100	2509

The off-site dose at unrestricted area due to release of the liquid effluents during normal operation using K-DOSE60 was calculated. The results were compared with the dose limit of MOST Notice 2001-2 and shown in Table 8.

Table 8. Results of off-site individual dose

Description	Result	MOST Notice 2001-2
Effective Dose (mSv/yr)	3.61E-03 (5 year-old)	0.03
Organ Equivalent Dose (mSv/yr)	4.10E-02 (Thyroid- 5 year-old)	0.1

CONCLUSIONS

The expected annual activity rate was calculated using the revised PWR-GALE (NUREG-0017, Rev.1). The released concentration at unrestricted area was evaluated using equation (1) and (2). The results of release concentration at unrestricted area for KORI 3&4 are presented in Table 4 and compared with the release limits of MOST Notice 2001-2, colume 8 in Table 3. And we confirmed that the release concentrations were less than its limits of MOST Notice 2001-2. As the results of off-site dose at unrestricted area using K-DOSE60, the effective dose was 3.61E-03 mSv/yr to the age of five and the organ equivalent dose was 4.10E-2 mSv/yr to thyroid of the age of five. From these results in Table 8, we also confirmed the off-site dose was within the limits of MOST notice 2001-2. Therefore, from the above results, it was confirmed that the liquid radwaste treatment system is maintained the performance of operation even though the fission products corresponding to 1% fuel failure is leaked into the system and that the release concentration and off-site dose at unrestricted area were well below the regulation limits of MOST Notice 2001-2.

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