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■ 기술논문 ■

Assessment of Post-LOCA Radiation Fields in Service Building Areas for Wolsong 2, 3, and 4 Nuclear Power Plants

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월성 원자력 발전소 2,3,4호기에서의 LOCA 사고후 보조건물의 방사선장 평가

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Abstract − The radiation fields following the large loss of coolant accident (LOCA) have been assessed for the vital areas in the service building of Wolsong 2, 3, and 4 nuclear power plants. The ORIGEN2 code was used in calculating the fission product inventories in the fuel. The source terms were based upon the activity released following the dual failure accident scenario, i.e., a LOCA followed by impaired emergency core cooling (ECC). Configurations of the reactor building, the service building, and the ECC system were constructed for the QAD-CG calculations. The dose rates and the time-integrated doses were calculated for the time period of upto 90 days after the accident. The results showed that the radiation fields in the vital access areas were found to be sufficiently low. Some areas, however, showed relatively high radiation fields that may require limited access.
Key words : Post-LOCA, ORIGEN2 Code, QAD-CG Code, Source Term, Radiation Field

요약 − 월성 원자력발전소 2,3,4 호기의 보조건물 주요 지역에서 냉각재 다량상실사고 (large LOCA) 후의 방사선장을 평가하였다. 핵분열 생성물의 총량은 ORIGEN2 코드를 사용하여 계산하였고 선원항은 2중고장 시나리오, 즉 LOCA 사고후 비상노심냉각 (ECC) 계통의 고장이 결부된 사고시의 방사능 방출에 근거하였다. 원자로건물, 보조건물 및 ECC 계통의 구조모형을 QAD-CG 모델에 포함하여 계산하였다. 사고시점부터 90일 경과시까지 시간대 별로 선량율과 누적선량을 계산하였다. 결과적으로, 연속출입이 요구되는 중요지역에서의 방사선장은 충분히 낮은 것으로 평가되었다. 그러나, 일부구역에서는 제한적인 출입을 허용할 정도로 상대적으로 높은 방사선장을 나타내었다.
중심단어 : 냉각재 상실사고, ORIGEN2 코드, QAD-CG, 선원항, 방사선장

INTRODUCTION

The large LOCA in CANDU reactor is the loss of primary circuit cooling resulted from the loss of heavy water coolant due to a break or leak in the primary circuit, such as the reactor inlet/outlet head or pump suction pipes. Even after the large LOCA which releases the radionuclides, certain areas such as main control room (MCR) and secondary control area (SCA) should be accessible continuously in order to perform the appropriate safety work as required by the accident management actions. There are also some othre areas requiring the access of relatively short time period for the proper safety work.

The purpose of this study is to assess the radiation fields in the MCR, SCA, and various areas of the service building following the accident. The radiation levels were calculated from zero to 90 days after the accident, and this result would provide the guidance in allowing the access into such areas to perform the safety work. The Wolsong 2, 3, and 4 shielding design criteria to protect the plant operation personnel under the postulated accident conditions is as follows:

The dose over a ninety day period following an accident should be less than 100 mSv(10 rem) in an area of continuous occupancy. The limit of 100 mSv was adopted on the basis of ICRP-40[1].

In this analysis, the identification of those areas in which the external fields exceeds 2 mSv/h was also made in accordance with the shielding design guidelines for CANDU-6 reactor.

The source terms in this analysis are based upon the dual failure accident of the LOCA coincident with the impairments of emergency core cooling system (ECC system), which was chosen as the bounding accident scenario. The release amounts of radionuclides from this accident are taken from the Wolsong 2 preliminary safety analysis report (PSAR)[2], in which the analysis was performed based on the 24 % reactor inlet head break. The isc*ope generation and depletion code ORIGEN2[3] was applied in fuel reprocessing mode in which the release amounts of each fission product were taken out of the total core inventory and processed in decay mode from time zero to 90 days after the accident.

The reactor building, whole service building structure, and the ECC system including the ECC pumps, pipes, and heat exchangers were incorporated into the single QAD-CG[4] model, in which they were represented as source containers as well as shielding structures. The QAD-CG calculated the pseudo dose rates based on the arbitrarily gamma source strength of 1.0×10^{10} gammas/sec-cm^3 of each 11 energy groups which is compatible with the energy groups of actual fission product gamma sources calculated by ORIGEN2. Finally, the actual dose rates and integrated doses were obtained by using the utility program EQ which normalizes the QAD-CG pseudo dose rates to the actual gamma sources by the ORIGEN2 results.

SOURCE TERM CALCULATION FOLLOWING LOCA WITH ECC IMPAIRMENTS

Among the 834 radioactive fission products and actinides in the irradiated fuel, only 18 elements were considered to be important radiation sources. These elements were categorized to seven release groups, i.e. noble gases(Xe and Kr), halogens(I and Br), alkali metals(Rb and Cs), alkali earth metals(Sr and Ba), noble metals(Mo, Tc, Ru, and Rh), and rare earth metals(Y, Ce, Pr, and Eu)[5, 6].

If LOCA occurs, the ECC system, which is designed to perfrom high pressure injection followed by medium pressure and subsequent low pressure injection, would be initiated. The impaired ECC system means the malfunction of high pressure injection system, but the medium pressure injection system is still available to operate. During the medium pressure injection, the ECC system pumps located in the service building take suction from the dousing tank and discharge the dousing tank water to the reactor headers via ECC pipings for the appropriate cooling of the heat transport system. The pump discharge to the reactor headers passes through the ECC heat exchanger. Among the important radionuclides, the noble gases released from the fuel escape easily from the core and exists in airphase in the reactor building containment. Except for the noble gases, most of the iodine and other elements were considered to exist in waterphase in heat transport system and ECC system, which provide the waterborne radiation sources[6].

Determination of Activity Releases Following the Accident

The amount of radionuclides was taken from the Wolsong 2, 3, and 4 PSAR which assumed the 24 % reactor inlet head break where the LOCA with ECC impairment accident scenario was applied. Figure 1 shows the bound inventory of I-131 release from fuel as a function of time after blowdown for

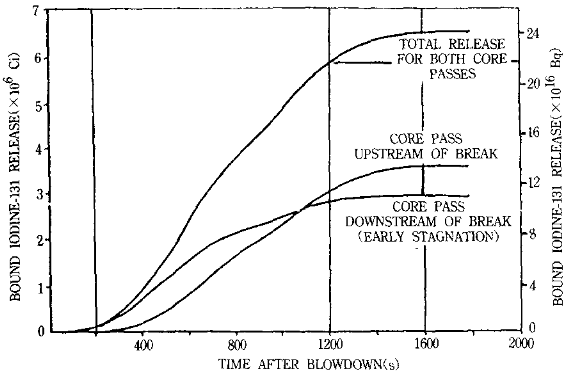

Fig. 1. Iodine-131 releases in each core pass for broken loop (5 g/sec steam flow)

a steam flow rate of 5 g/sec which was found to give the maximum fuel temperature [2], and this amounts to about 2.4×10^5 TBq of iodine activity. The free gap inventory releases are about 0.1×10^5 TBq. The total for both the grain boundary and free gap inventories for I-131 thus, amounts to about 2.5×10^5 TBq, and this is about 10.8 % of the total core inventory for I-131. This was taken to be released over the three time intervals shown in Figure 1, viz., 0.3 % between time zero and 200 sec, 9.4 % between time 200 sec and 1200 sec, and 1.1 % between 1200 sec and 1600 sec.

Table 1. Fission products cosidered for source term calculations by ORIGEN2

Category	Fission Porducts
Halogens	I, Br
Noble Gases	Xe, Kr
Alkali Metals	Rb, Cs
Alkali Earth Metals	Sr, Ba
Tellerium Group	Te, Sb
Noble Metals	Mo, Tc, Ru, Rh
Rare Earth Metals	Y, Ce, Pr, Eu

Table 2. Release amounts of fission products as a percent of its total core inventory incorporated into ORIGEN2

Run #	Element	Air Phase (%)	Water Phase (%)
1	Kr, Xe	10.8*	0.0
2	I, Br	0.108	10.8*
3	Rb, Cs,Te, Sb	0.108	10.8*
4	Sr, Ba	0.016	1.60*
5	Mo, Tc, Ru, Rh	0.0032	0.32*
6	Y, Ce, Pr, Eu	0.0017	0.17*

* Release amounts incorporated into ORIGEN2

It was assumeed that 10.8 % of the total core inventory of noble gases was released. This assumption overestimates the noble gases releases of 2.4×10^5 Bq.J, which amounts to about 8 % of the total core inventory, reported in Reference 2. The percentage of the core inventory release for those elements (except for I, Br, Xe, and Kr) was taken to 10.8 % times the ratio of the fractional releases from fuel for a melt-type accident between that element and iodine[5 and 6]. For these elements, the same percentage as the I-131 was applied in determining the release fraction over the three time intervals. The fission products and their release amounts are given in Tables 1 and 2.

ORIGEN2 Calculation

Based on the release amounts of each element, six ORIGEN2 calculations were performed in the fuel reprocessing mode to come

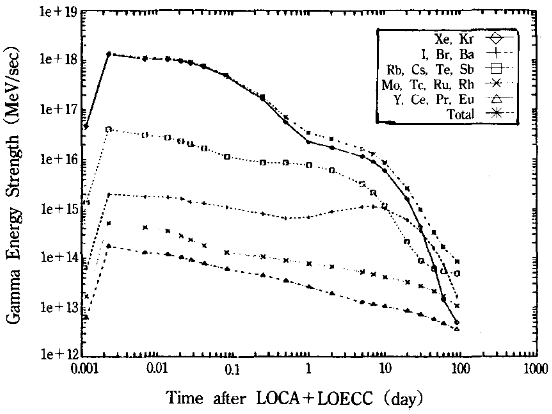

Fig. 2-1. Airborne gamma source strength following LOCA+LOECC

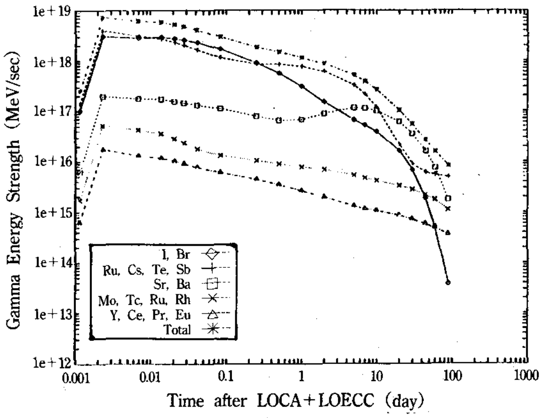

Fig. 2-2. Waterborne gamma source strength following LOCA+LOECC

up with fission products gamma sources. The calculations of fission product inventories were based on the average bundle fission power of 473 kW irradiated to a mean burnup of 350 MWD/MTU. Following this irradiation, the amounts of each radioactive elements were recovered and released over the three time intervals, short, intermediate, and long term. The decay of all perents and daughters produced were followed by the decay chains of the ORIGEN2 code, and the gamma source strength of six groups were obtained in MeV/sec. Figure 2-1 and 2-2 show the airborne and waterborne fission product gamma source strength of each group based on the release fraction given in Table 2.

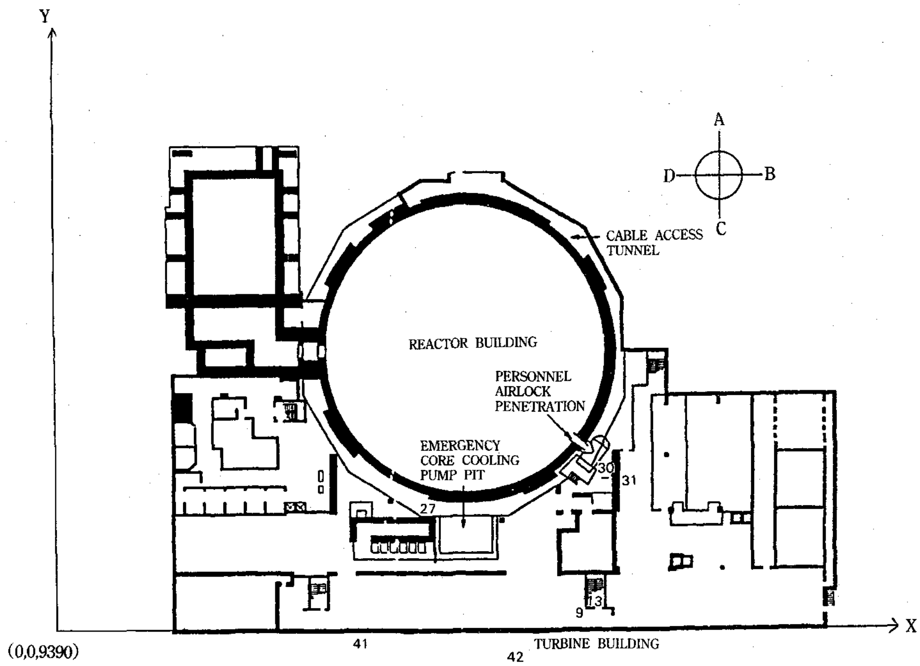

Fig. 3-1. Service building plan view (elev. 93.90 m)

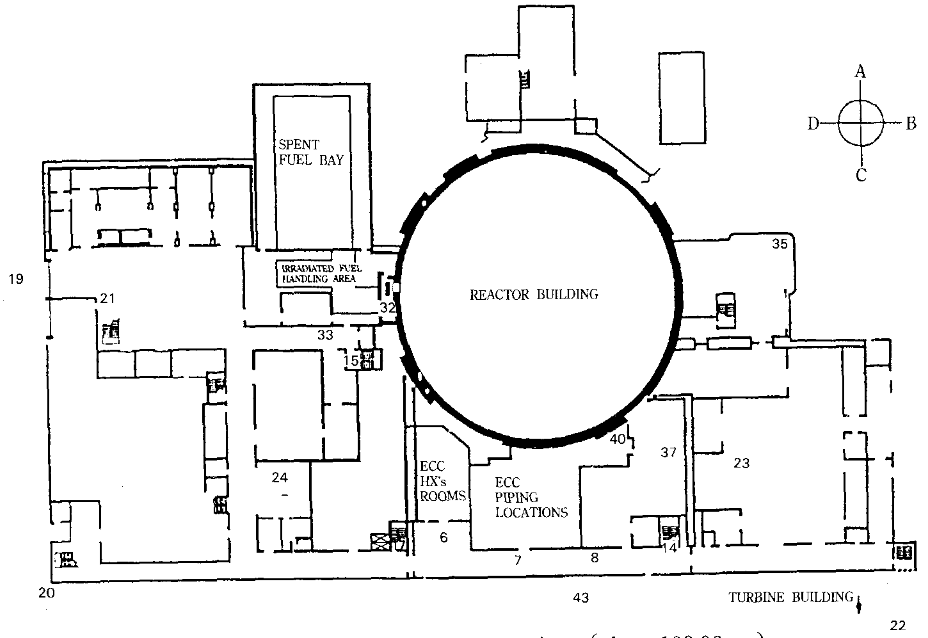

Fig. 3-2. Service building plan view (elev. 100.00 m)

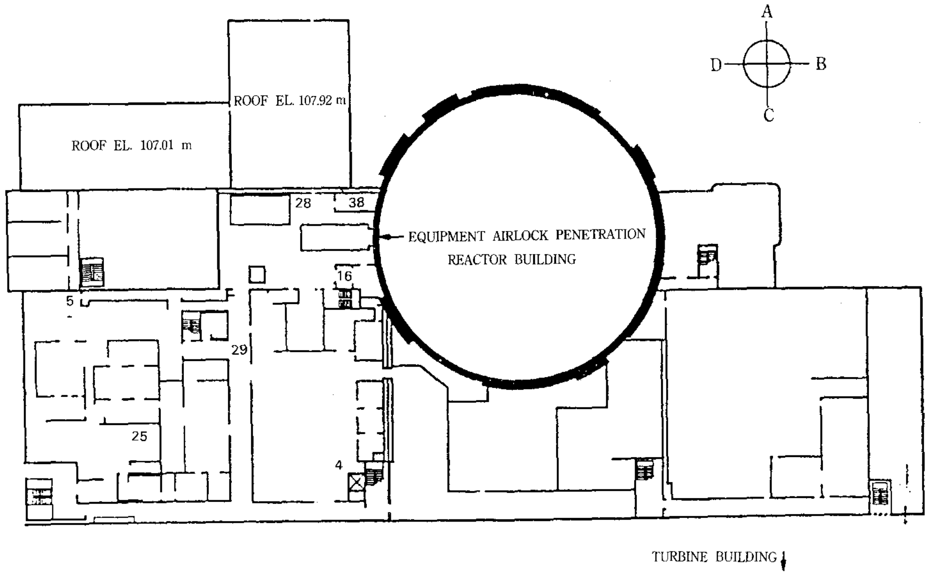

Fig. 3-3. Service building plan view (elev. 105.41 m)

QAD-CG MODELLING AND PSEUDO DOSE RATES CALCULATION

The point kernel code QAD-CG, combinatorial geometry version of QAD-P5A code, was used to model the reactor and service building structure from the base elevation up to roof. The reactor building and service building structure are given in Figures 3-1 to 3-4 from the base elevation up to roof. The QAD-CG model includes reactor building containment with its airborne sources, service building floors with shielding structures such as concrete walls, ceilings, etc[7]. The model

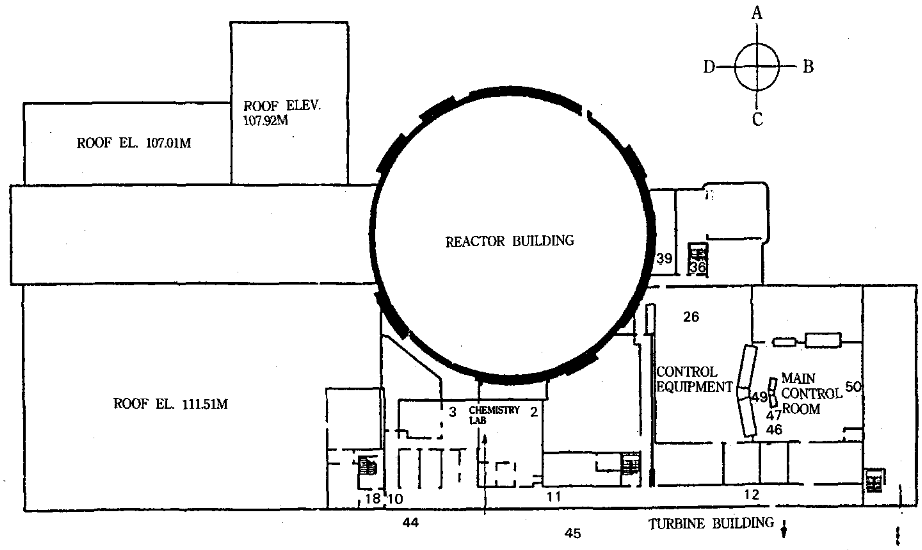

Fig. 3-4. Service building plan view (elev. 109.10 m)

also includes the ECC system heat exchangers, pumps, and all the pipings filled with water[7], which give the waterborne radionuclide gamma sources following the accident. More than three hundred bodies and regions were defined in the model.

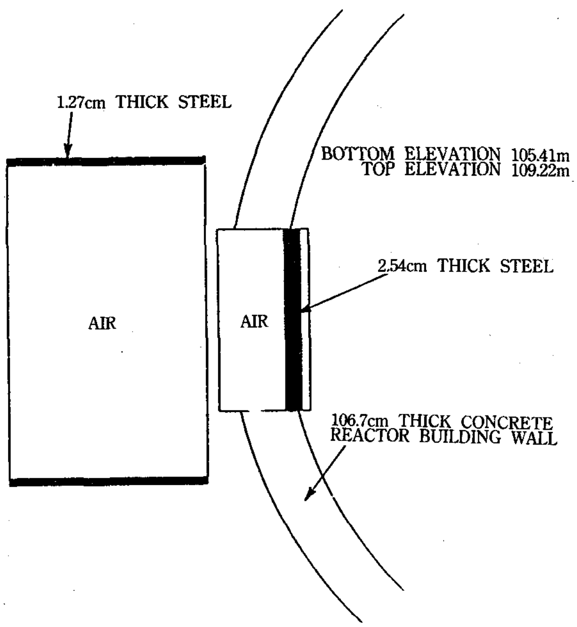

Fig. 4-1. Modelling of equipment airlock penetration by the QAD-CG code

Modelling of Reactor Building with Its Airborn Suurces

The reactor building was modelled as a cylinder having 2072.6 cm inner radius and 2179.3 cm outer radius, and 3557.0 cm equivelent height, which gives the total free air space volume of 48,000 m^3, and this is taken from Appendx VI in Chapter 15 of Reference 2. The inside of reactor building structure was neglected, and it was assumed to be filled with air. The 106.7 cm thich concrete containment provides the major shielding of the airborne gamma radiation which would be transported to service building. The main airlock and personnel airlock were incorporated into the containment modelling so as to account for the weak shielding effect from these. The modelling of main airlock penetration and personnel airlock penetration are shown in Figures 4-1 and 4-2, respectively [7]. The airborne sources were assumed uniformly distributed inside the reactor building. To represent the reactor building sources in reasonable point kernel sources, the radial mesh size is decreased as it goes to outward direction with uniform mesh increments along axially and azimuthally.

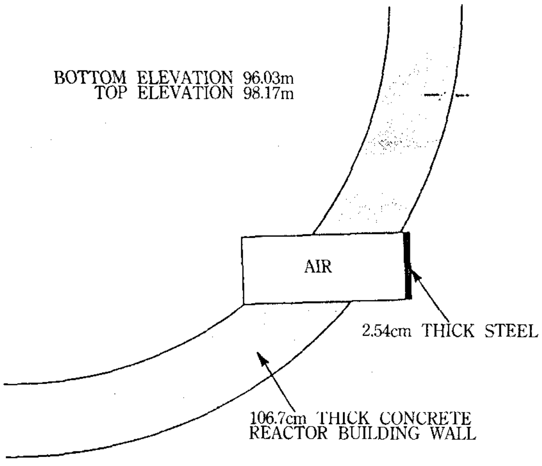

Fig. 4-2. Modelling of personnel airlock penetration by the QAD-CG code

QAD-CG Modelling of ECC System with Its Waterborne Sources

The ECC system sources consist of two heat exchangers, two pumps, and various pipings. The ECC heat exchangers are plate types which have the radioactive primary liquid volume of 1158 m^3 and secondary liquid volume of 1158 cm^3, which is not active. Therefore, volume of 1158 cm^3 was used to represent the heat exchanger source volume. These heat exchangers are homogenized based on the weight and volume of its components such as water, steel plate, etc., to provide the equivalent attenuation coefficients. 16W, 12W, and 18W types of ECC pipings were represented as square pipes which have the equivalent wall thickness and cross sectional areas. The ECC system sources were broken down into 23 sources as shown in Figure 5. Figure 5 and Table 3 show the modelling of the ECC system with the piping layout and the associated information, respectively.

Choice of Dose Point Locations

Dose point locations were chosen as follows[8].

1) Areas requiring continued access:
Continued access to the main control room

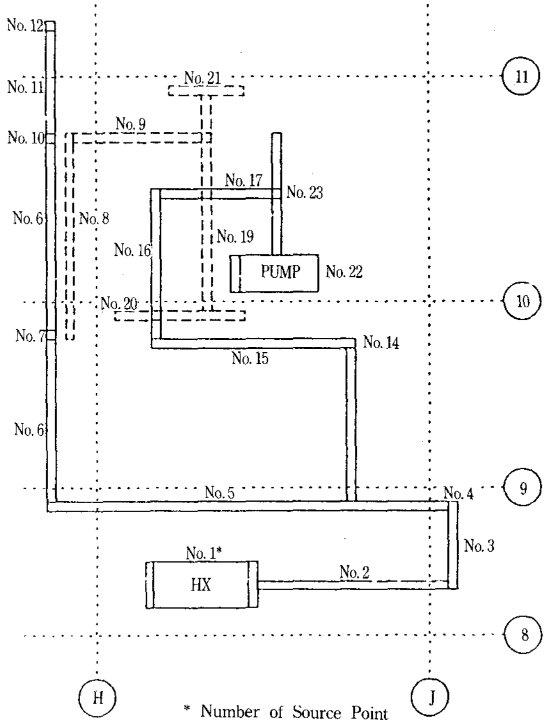

Fig. 5. QAD-CG code modelling of ECC system

Table 3. QAD-CG representation of the ECC system HXs, pumps, and piping*

Source No.	Actual Dimensions (cm)	Model** Dimensions (cm)
1	106×141×220	106.0×141.0×220.0
	215.0	215.0
2	215.0	
	405.1	471.7
3	471.7	
	412.5	528.3
4	300.7	
	948.3	948.3
5	1140.0	1140.0
6	680.0	
	490.2	1139.9
7	68.6	
	504.3	
	650.	650.0
8	320.	320.0
9	558.8	833.1
10	274.3	
	590.0	590.0

(to be continued)

Source No.	Actual Dimensions (cm)	Model** Dimensions (cm)
11	770.1	770.1
12	850.	850.0
13	172.8	172.8
14	543.6	543.6
15	392.5	392.5
16	340.4	340.0
17	132.0	355.9
18	97.5	
	223.9	
	55.6	
	712.5	712.5
19	712.5	
	294.0	294.0
20	294.0	294.0
21	100.2×100.2×332.4	100.2×100.2×332.4
22	446.0	446.0
23	446.0	446.0

* Cross sectional area for each pipe of 16 inch, 12 inch, and 18 inch is 1178.8 cm^2, 740.23 cm^2, and 1486.12 cm^2, respectively

** Refer Figure 5

is required. Fields in the secondary control area were reported lower than the main control room.

2) Access routes:

Fields on routes to and from the areas requiring continued access and throughout the service building were assessed.

3) Assembly areas and work areas:

Fields at relatively short times after accidents are of most interest in areas such as the administration building area, crane hall assembly area, turbine building southeast area, mechanical maintenance shop, electrical instrumentation and control shop, and chemistry lab.

4) Containment isolation locations:

Fields were assessed at instrument, service building, and breathing air isolation locations, and at containment isolation points of other auxiliary systms that may require isolation.

5) Airlock seal air supply:

Fields were assessed at the original locations of the N_2 bottle quick-connects for the equipment airlock, for the personnel airlock, and for the spent fuel discharge bay containment door. (These quick-connects would be used before time 24 hours to ensure airlock seal air supply at times greater than 24 hours.)

6) Sampling areas:

Fields were assessed at the stack sampling area, and at the primary heat transport system and moderator sampling area.

7) Filtered air discharge system area:

Fields at the filter, for accidents not involving containment of the filter, were evaluated.

8) Other areas:

Those include several turbine building access areas, fire water pump house, and the main parking lot.

Detailed locations of the dose points with specific information are shown in Table 4.

CALCULATION OF DOSE RATES AND INTEGRATED DOSES

The QAD-CG calculated the pseudo dose rates from the waterborne and airborne sources at 50 dose points located throughout the plants. Both airborne and waterborne calculations were based on the arbitrarily source strength of 1.0×10^{10} gammas/sec-cm^3 of each 11 gamma energy groups. The concrete buildup factor was used in the overall calculations, which is built-in in QAD-CG.

The EQ program normalizes the pseudo dose rates by the actual source terms calculated by ORIGEN2. Since the ORIGEN2 results were based on the one bundle power (473 kW), the normalization were also made by the reactor fission power (2158.5 MW) to get the dose rates/doses from the total core inventory sources. The program also accounts

Table 4. Dose points locations* in QAD-CG modelling

Dose Point	X Axis Coor.	Z Axis Coor.	Y Axis Coor.	Description
1	−229	10761	3252	Administration Building
2	745	11031	1402	Chemistry Lab
3	6042	11031	1402	Chemistry Lab
4	4572	10661	823	Electrical Shop
5	762	10661	3140	Change Area
6	5816	10122	305	Corridor S-148
7	7048	10122	284	Corridor S-148
8	8055	10122	284	Corridor S-148
9	9080	9511	244	Corridor S-009
10	5467	11031	107	Corridor S-302
11	7902	11031	107	Corridor S-302
12	11109	11031	107	Corridor S-302
13	8217	9511	275	Stairwell S-010
14	9217	10121	244	Stairwell S-010
15	4572	10121	3140	Stairwell S-002
16	4925	10517	3201	Stairwell S-002
17	5208	10121	427	Stairwell S-007
18	5178	11031	82	Area S-323
19	−10	10151	4344	Fire Water Pumphouse
20	−3048	10151	−4115	Parking Lot
21	762	10151	3993	Crane Hall
22	12252	10151	−5761	Turbine Building South-east Area
23	10347	10151	1646	Mechanical Maintenance
24	3429	10151	1259	Radiation Protection Equipment Storage
25	1524	10661	1128	Lockers Area
26	10103	11001	2881	Control Equipment Room NE Corner
27	6706	9793	1646	Compressed Air Isolation
28	3932	10608	3231	Equipment Airlock Area
29	3179	10608	2881	Equipment Airlock N_2 Quick Connects Location
30	9361	9451	2317	Personnel Airlock Area
31	9589	9451	2073	Personnel Airlock N_2 Quick Connects Location
32	4950	10091	3676	SF Bay Door Area
33	4191	10091	3584	SF Bay Door Seal Quick Connects Location
34	8253	10091	1235	RBVS Exhaust Filter
35	10286	10091	4343	H_2 Isolation Panel
36A	10175	10091	3836	Stairwell S-017
36B	10175	11051	3836	Stairwell S-017
37	9104	10091	1646	D_2O Sampling Cabinet
38	4700	10603	4532	Resin Transfer Room
39	9724	11062	3643	Stack Sampling
40	1798	10091	8268	Dampers 3831-DP22
41	5778	9527	−1235	CI III and CI IV Switchgear Access
42	9561	9527	−2058	General TAB Access

(to be continued)

Dose Point	X Axis Coor.	Z Axis Coor.	Y Axis Coor.	Description
43	7932	10121	−1341	General TAB Access
44	5816	11028	−884	Switchgear & MCC Access
45	8253	11028	−1235	General TAB & Reserve FW Access
46	11490	10944	1235	MCR General Field
47	11490	11120	1235	MCR General Field
48	11490	11031	1555	MCR Opposite CER Floor Slots
49	11490	11299	1555	MCR Peak Field Area
50	12542	11044	1646	MCR General Field

* Refer Figure 3-1 for coordinate origin.

for the actual step by step release during a given time interval, whereas the ORIGEN2 assumed that the release occurred at the start of a given time interval. For the dilution of activity in the containment atmosphere, a free air space volume of 48,000 m^3 was used. It was assumed that the medium pressure ECC is operating, hence the whole dousing tank volume can be available for the ECC system. The total amount of water available was taken to be the coolant volume from the heat transport system (about 160 m^3) plus the dousing tank water volume of 2060 m^3, a total of about 2220 m^3.

Table 5. Dose rates and integrated doses for main control room (dose point #49)

Time After Accident (day)	Dose Rates from R/B (mrad/h)	Dose Rates from ECC (mrad/h)	Integrated Doses from R/B (mrad)	Integrated Doses from ECC(mrad)
0.000E+0*	0.000E+0	0.000E+0	0.000E+0	0.000E+0
1.157E−3	5.340E−1	2.264E−3	7.414E−3	3.144E−5
2.314E−3	8.630E−1	3.138E−3	2.681E−2	1.064E−4
6.944E−3	5.955E+0	1.409E−2	4.056E−1	1.063E−3
1.389E−2	1.052E+1	1.068−2	1.779E+0	3.578E−3
2.083E−2	1.058E+1	1.297E−2	3.536E+0	5.998E−3
2.778E−2	9.769E+0	1.022E−2	5.233E+0	7.933E−3
4.167E−2	8.386E+0	6.861E−3	8.259E+0	1.078E−2
8.333E−2	5.575E+0	2.917E−3	1.524E+1	1.567E−2
2.500E−2	1.718E+0	9.935E−4	2.982E+1	2.349E−2
5.000E−1	3.842E−1	6.790E−4	3.613E+1	2.851E−2
1.000E+0	2.666E−2	5.048E−4	3.860E+1	3.561E−2
2.000E+0	6.462E−3	4.856E−4	3.900E+1	4.750E−2
5.000E+0	6.486E−3	4.980E−4	3.946E+1	8.291E−2
7.000E+0	5.977E−3	4.602E−4	3.976E+1	1.059E−1
1.000E+1	5.070E−3	3.914E−4	4.016E+1	1.365E−1
2.000E+1	2.829E−3	2.190E−4	4.111E+1	2.098E−1
3.000E+1	1.626E−3	1.260E−4	4.164E+1	2.512E−1
4.500E+1	7.231E−4	5.601E−5	4.206E+1	2.840E−1
6.000E+1	3.244E−4	2.511E−5	4.225E+1	2.986E−1
9.000E+1	6.875E−5	5.304E−6	4.239E+1	3.095E−1

* read as 0.000×10^0

Table 6. Location of dose points exceeding 2 mSv/hour

Dose Point	Peak Dose Rates (mSv/hour)			Remarks
	Reactor Building	ECC System	Total	
2	9.4878E−01	3.9651E+01	4.0600E+01	Chemistry Lab
3	7.3657E−03	7.6486E+02	7.6487E+02	Chemistry Lab
4	3.8679E−02	1.0009E+01	1.0048E+01	Electrical Shop
6	3.3879E−03	4.8266E+02	4.8266E+02	Corridor S−148
7	1.2848E−01	1.1118E+02	1.1131E+02	Corridor S−148
9	1.1505E−01	3.0138E+03	3.0139E+03	Corridor S−148
11	2.1571E−01	7.0757E+00	7.2914E+00	Corridor S−148
13	5.1626E−03	2.7071E+03	2.7071E+03	Stairwell S−010
16	3.3251E+00	4.0824E+01	4.4149E+01	Stairwell S−010
19	2.9880E+01	1.8578E−02	3.9899E+01	Fire Water Pump
21	6.5780E−01	2.9419E+00	3.59997E+00	Crane Hall
23	1.0754E−03	1.0123E+01	1.0124E+01	Mech. Maintenance
27	6.0483E+00	5.7028E+04	5.7034E+04	Compressed Air Isolation
28	3.5432E+01	2.9266E−06	3.5432E+01	Equipment Airlock Seal
30	3.0333E+01	2.6876E+03	2.7179E+03	Personnel Airlock Area
32	5.4450E+01	1.2206E−03	5.4451E+01	SF Bay Door Area
33	7.5588E+01	3.5489E−04	7.5588E+01	SF Bay Door Seal Quick Connects Location
34	2.1780E−01	2.5278E+02	2.5300E+02	REVS Exhaust Filter
35	3.9518E+00	3.0596E−11	3.9518E+00	H_2 Isolation Panel
37	2.8510E+00	7.1870E+01	7.4721E+01	D_2O Sampling Cabinet
38	2.4657E+02	1.3709E+02	3.8366E+02	Resin Transfer Room
39	7.6991E+00	2.3668E−04	7.6993E+00	Stack Sampling
41	2.7178E+05	2.1054E+00	2.1054E+00	Switchgear Access
43	7.3883E+03	2.2287E+00	2.2361E+00	General TAB Access

RESULTS AND DISCUSSIONS

The dose rates and integrated doses at the main control room, where the continued access is required, were sufficiently low to allow such access. The dose rates and integrated doses in this area are shown in Table 5. However, relatively high fields were also found at several locations in which the external radition fields exceed 2 mSv/hour (200 mrem/hour) as identified in Table 6. The access to these areas after the accident should be controlled in order for the occupational radiation exposure to be maintained under the limit specified in the regulation.

In this calculation, it was assumed that the airborne activity would remain in the reactor building after the accident. If the reactor building ventillation system (RBVS) is used (which is normally used 24 hours after the accident), the activity in the containment is to be passed through filter and finally discharged into atmosphere by RBVS. In the course of this filtered air discharge, the airborne activity will be deposited in the filter and some will be distributed in the RBVS ducts located in the service building, and which extends to the stack. This would result in the redistribution of the airborne sources. Further study, therefore needs to assess the effect of RBVS, which will include the modelling of RBVS and source redistributions. It was also reported that the containment leaks some of its airborne radionuclides into service building within 24 hours after the accident. The leakage source also needs to be considered.

CONCLUSION

Radiation fields have been determined for 50 locations throughout the plant for the dual failure accident scenario, i.e., LOCA coincident with impaired ECC system. Radiation fields have been calculated at times ranging from zero to 90 days after the accident. Dose rates above 2 mSv/hour are calculated to occur at 24 dose points. However, radiation fields are generally much lower than this, and areas requiring continued access remains sufficiently low to remain habitable for such access.

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