Development of a New Clearance Monitor for Exemption of NPP Radioactive Wastes and its Performance Test - 17200

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ABSTRACT

In nuclear power plants, it is common practice to clear radioactive wastes for free release by using clearance monitor with plastic scintillation detectors so as to reduce disposal costs. These monitors are possible in part thanks to the excellent detection levels achievable with high energy and high yield photons emitted from contaminants such as Cs-137 and Co-60. However, it is difficult for the monitors to separate Naturally Occurring Radioactive Materials (NORM) from the radioactive wastes because the plastic scintillation detectors have poor resolution compared to NaI or HPGe detectors. KHNP has developed a new clearance monitor for determining whether or not the radioactivity level of metal scraps satisfies regulatory criteria. The monitor is composed of 12 large area plastic scintillation detectors and 3 NaI scintillation detectors. The measuring chamber is surrounded by a 5cm-thick lead shield with chamber dimensions of 1.5m x 1.0m x1.0m. Its algorithm has a function of measuring total gamma activity while analyzing gamma nuclides and identifying 27 hotspots in radioactive wastes. The detection limit of this system, according to the MCNP simulation, was determined to be about 45 Bq for Co-60 and 78 Bq for Cs-137 with the condition of an empty chamber and measuring time of 1 minute. We evaluated the self - shielding effect of waste for background correction factor and waste height (volume in the chamber) correction factor. We also tested its MDA, location determination of hot-spots, temperature compensation function and convenience of operation. In the future, we will plan to install it at a nuclear site for more tests including evaluating its durability. [1] [2]

INTRODUCTION

Radioactive waste below VLLW(Very Low Level Waste) but above EW(Exempt Waste) criteria amounts to about 2500 tons as of 2011 and is stored at nuclear power plants in Korea. Of this, roughly 620 tons are metal scrap which is mostly composed of the machinery in air conditioners, pipes, galvanized sheets, ducts and valves/motors and other machines. Among these, the flat plates, H-beams and other waste which is easy to measure its radioactivity and decontaminate are mostly applied for permission of clearance. The main purpose of the clearance monitor is to complete an analysis of the representative samples to re-confirm the wastes that were evaluated as suitable for clearance before they are finally

released to the environment. It was designed and manufactured to measure the total gamma radiation in a short period of time.

The result of the radioactivity measurements for clearance performed during the last 3 years showed that no radioactive nuclide in the waste was found to be detected except waste oil whose activity was determined to be less than MDA. And the MDA was about 10^{-3} Bq/g, which is 1/100 of the clearance limit (Cs-137, Co-60 : 0.1 Bq/g).

The Korean 'regulations regarding the categorization and clearance criteria of radioactive waste' are similar to the standards defined by IAEA, as it categorizes radioactive wastes into 4 levels – high level waste(HLW), intermediate level waste(ILW), low level waste(LLW) and very low level waste(VLLW); it categorizes clearance waste as a general waste, which is evident according to regulation of the radioactive waste.[3] However, when looking at how the clearance procedure is actually being implemented, there should not be a detection of radioactivity that is at least 1/10 higher than the lower boundary detection conditions in terms of the concentration of the radioactivity for each nuclide of the IAEA-clearance level. Also, in the radiation exposure assessment according to the clearance scenario, it should be proven that the yearly exposure dose must be less that 10 μSv. Thus, the procedure is being used very conservatively.

RESULT AND DISCUSSION

1. Review of performance improvements to be required for the current clearance monitors

It was discovered that the weight measurement device (load cell) of the wastes is inaccurate when it is less than 100 kg. Also, the shielding doors, which are opened and closed in a top to bottom direction, have a severe vibration issue and a lengthy activation time that need to be improved.

Also, the results of the test using the calibration sources show that radiation measurement is underestimated by 64%, and that calibration of the adjustment efficiency of the density/volume/shape of the wastes is necessary.

2. Determination of the MDA

The MDA of the radioactivity detector is related to background, measurement time, and other factors. The background of the chamber of the clearance monitor is affected by cosmic rays, terrestrial radiation, radon decay materials in the air and others. However, not only is each amount different for each region, but the number of radioactive nuclides is so large that it is difficult for a computer simulation to calculate them. Therefore, the background was measured using the

conditions, based on actual measurement of the chamber size and the shield thickness in the lab. The interior of the chamber of the clearance monitor was simulated for the gamma ray energy spectrum and the response time of Co-60 and Cs-137. The sources were placed and measured by plastic detectors in the center of the chamber, which was the area that was estimated to have the highest MDA. The simulation of the MDA was carried by using the counting rate of background and energy spectrum of Co-60, Cs-137.

After using a miniature chamber (spatial gamma dose rate: 120 nSv/h) to install a 3” x 3” x 3” plastic detector within the 5 cm thick shielding board, (#1, #2) in order to use the MCA to calculate the total counting number of the spectrum, the result showed that as the chamber volume increased, the counting number of the background increased as well.

Table 1. Measurement of Background Radiation

Size of Chamber			Volume of chamber (cm^3)	Shielding (cm)	Counting Rate(cps)
W(cm)	D(cm)	H(cm)			
10	40	10	4,000(#1)	5	10.3 ± 0.05
30	40	20	24,000(#2)		11.1 ± 0.05

The size of the measurement chamber was set into two forms, as shown in figures 1 and 2. Also, the simulation of the spectrum was done using the MCNP code to analyze the Co-60 and Cs-137 that are in the middle of the chamber.

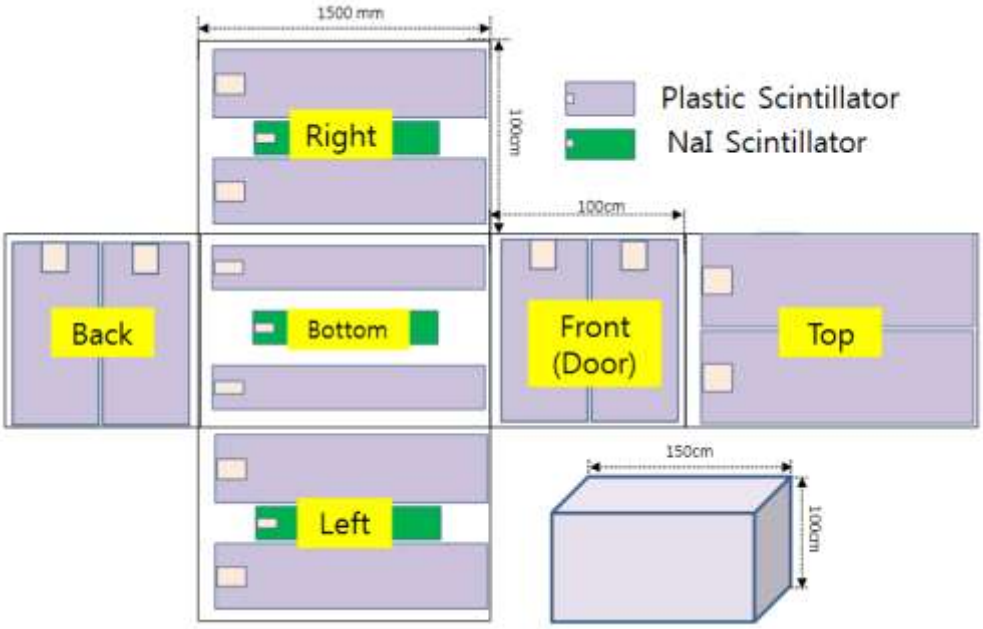

Figure 1. Measurement Chamber (150 cm × 100 cm × 100 cm)

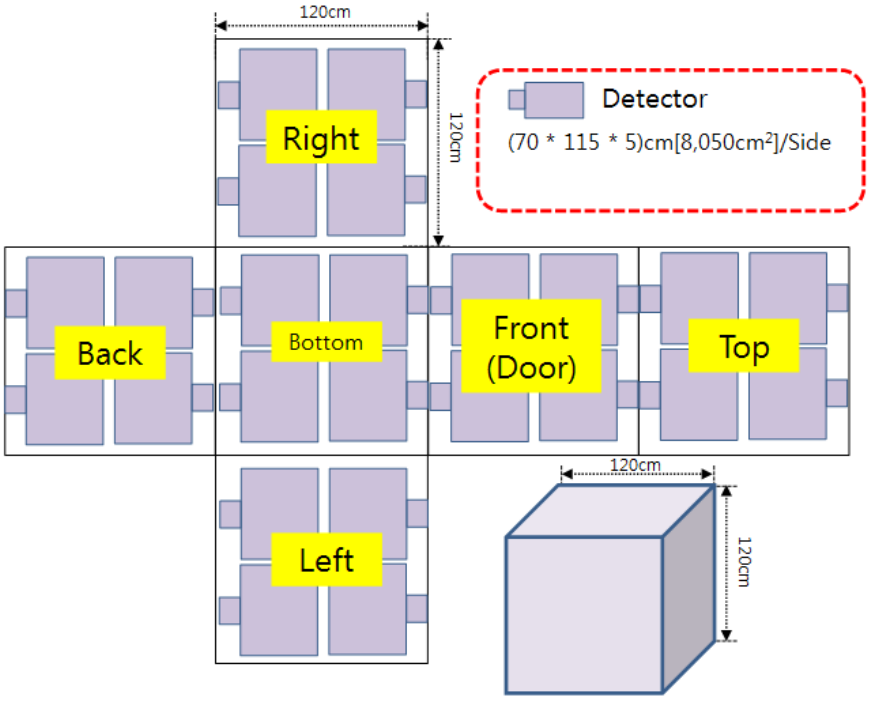

Figure 2. Measurement Chamber (120 cm × 120 cm × 120 cm)

Metal scraps were cut into piece (maximum 150 cm) and held in storage in forms that had side lengths of 150cm, as shown in figure 1. In this case, the area of

Chamber 1 for each side (for top/bottom/left/right sides) was 105 cm x 100 cm x 5 cm, and the two front/back sides were 60 cm x 100 cm x 5cm. In case of the measurement chambers that were in cubic form, as in figure 2, all six sides had an area of 70 cm x 115 cm x 5 cm. Table 2 shows the chamber volume, detector volume and area for each simulated condition.

Table 2. Size of chamber and detector for each simulated condition

Condition	Chamber Size(m^3)	Size of detector	
		Volume (m^3)	Area (m^2)
Condition#1	1.5	0.27	5.4
Condition#2	1.7	0.24	4.8

The calculation of the MDA was done by using background measurement data, which was measured in the laboratory and by MCNP simulation to effectively use the total efficiency for each detector. Also, the MDA was measured for each weight by assuming that the density within the chamber filled with iron is 1000 kg/m3. The result can be seen in Table 3.

Table 3. Result of MDA simulation

Condition	Nuclides	MDA Test Result	
		Air, Empty Chamber (Bq)	Iron (Bq/kg)
Chamber #1	Co-60	44.7	0.681
	Cs-137	77.7	4.19
Chamber #2	Co-60	44.9	0.948
	Cs-137	83.5	6.725

3. Design and development of the clearance monitor

The clearance monitor was designed in detail according to the results of the MDA computer simulations. The main characteristics of the design are as follows: First, the shape of the wastes from the nuclear power plants was taken into consideration for design of the quadrangle shape of the chamber. Second, in order to adapt effectively to malfunctions that could occur during operation, it was maneuvered to automatically withdraw the detecting element when needed. Third, the insensitive area inside the chamber was minimized, as roughly 71% of the chamber area was covered with plastic scintillation detectors. Fourth, in order to accurately measure the total gamma radiation of the target wastes, and to identify the presence of natural radioactivity and main nuclides, twelve plastic scintillation detectors and three high-efficiency NaI scintillation detectors were added.

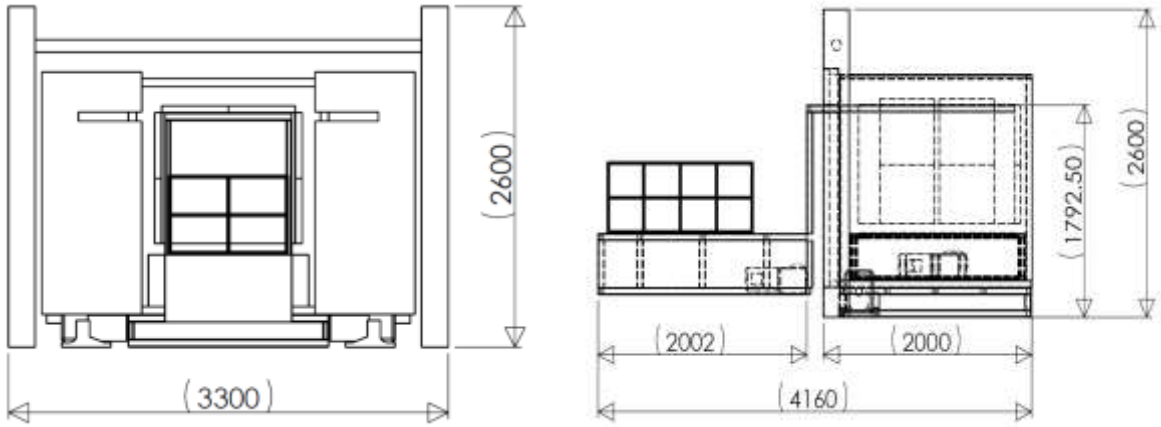

Figure 3. Design of device for clearance monitor

The clearance monitor has parts with the following functions: The driving part has the function of moving the sample container from the sample load location to the measurement location, or from the measurement location to the sample load location. During the movement of the sample load, the door of the measurement chamber interlocks to open and close. The signal unit measures the sample's temperature and weight, and controls 15 radioactivity detectors to the motion of the spectrum driving part(based on the radiation signals) that is related with the signal unit of the sensors. The algorithm determines exact location of contaminated waste in 27 regions in the chamber. It is also functional in measuring the radiation concentration of the sample. Lastly, the S/W of the PC, which operates all monitoring devices, is designed to adequately unify and operate all of the aforementioned functions.

4. Performance Test

The completed prototype was installed at the CRI to do two types of performance tests. The first was to locate the elements in 27 locations of the empty sample container, as well as to identify the radionuclides. Another usage was to charge and test the metal scraps and measure elements inside the sample container. For the metal scraps, a 220kg metal dummy was designed as seen in figure 4. The dummy is in a regular hexahedron shape with 50mm thick iron, and the inner parts are filled with Styrofoam. For a comparison test with the equipment available in the commercial market, the size and weight were designed to match the chamber size of the equipment currently used at NPPs. Figure 5 shows the location of the elements within the dummy, filled with Styrofoam. In this test, the elements used were Co-60 and Cs-137, and the radioactivity for each element was 6.74 kBq and 5.88 kBq respectively. They are 0.03 Bq/g and 0.02 Bq/g- 1/100, respectively, in comparison with the regulatory criteria (0.1 Bq/g).

Figure 4. Dummy for measurement

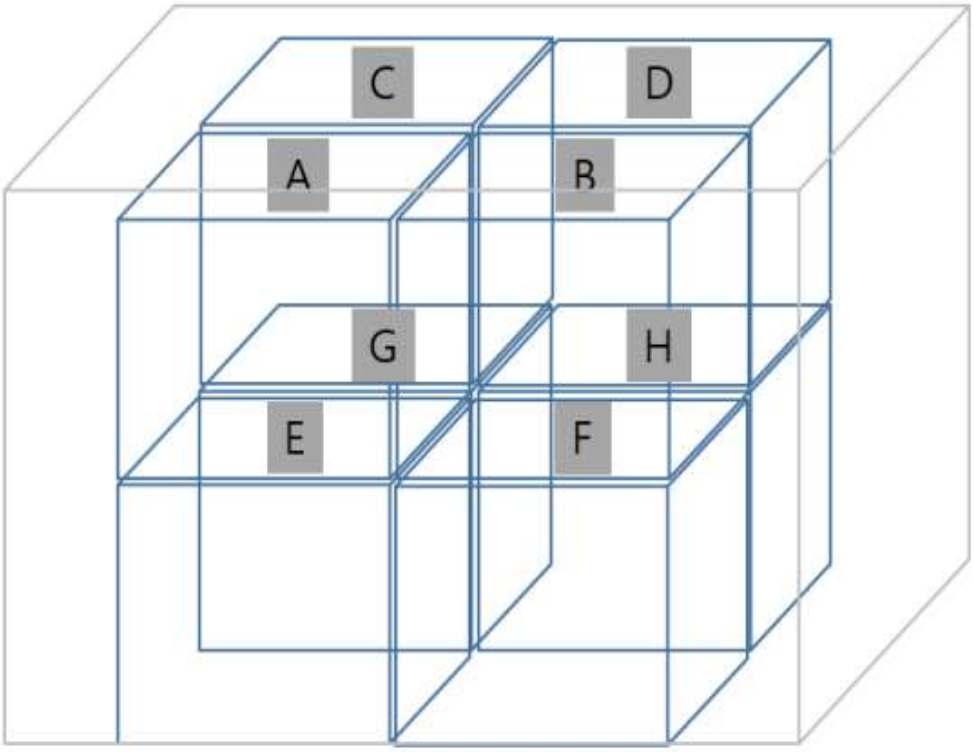

Figure 5. Location of the elements

The experiment results are shown in Table 4 and Table 5. Their radioactivity was measured within an error 6.3% and 12%. Identification of the elements' location was mostly accurate, and identification of the nuclides was done relatively accurately as well. Identification of the nuclides was done by finding the peak of the NaI detector spectrum and using the location of the peak (along with the energy), to identify the type of nuclide. If the detection sensitivity of the peak was high, it would identify even a false peak as a regular peak. If the detection sensitivity of the peak was low, it would not perform identification even for valid peak.

Table 4. Measurement result of empty sample containers (Co-60)

Location	Location for identification	Radioactivity (kBq)	Error (%)	Identification of nuclide
111	111	6.86	1.7	Co-60, Co-57, Mn-54
112	112	6.93	2.7	Co-60, K-40
113	113	6.78	0.5	Co-60, Co-57, Cs-137
121	122,222	6.60	-2.2	Co-60
122	122	6.71	-0.6	Co-60
123	123	6.79	0.6	Co-60
131	131	6.76	0.1	Co-60, Co-57
132	132,232	6.76	0.3	Co-60, K-40, Cs-137
133	133	6.72	-0.5	Co-60, K-40
211	211	6.80	0.7	Co-60
212	212	6.88	2.0	Co-60, K-40
213	213	6.79	0.7	Co-60, K-40
221	222	6.84	1.3	Co-60, Cs-134, Mn-54
222	222	6.78	0.5	Co-60, Cs-137
223	223	6.79	0.7	Co-60
231	231	6.80	0.8	Co-60
232	232	6.72	-0.4	Co-60
233	233	6.78	0.6	Co-60
311	311,312	6.69	-0.8	Co-60
312	312	6.83	1.2	Co-60
313	313	6.79	0.7	Co-60, K-40
321	322	6.63	-1.7	Co-60
322	322	6.81	1.0	Co-60
323	323	6.81	1.0	Co-60, Cs-137, Mn-54
331	331	6.75	0.1	Co-60, K-40, Co-57,
332	332	6.83	1.2	Co-60, K-40,
333	333	7.17	6.3	Co-60, Mn-54

Table 5. Measurement result of empty sample containers (Cs-137)

Location	Location for identification	Radiation (kBq)	Error (%)	Identification of nuclide
111	111	5.75	-2.3	Cs-137, K-40
112	112	5.71	-2.9	Cs-137
113	113	6.14	4.4	Cs-137
121	121	6.08	3.3	Cs-137
122	122	5.97	1.5	Cs-137, K-40, Co-57
123	123	6.50	10.5	Cs-137, K-40
131	131	5.31	-9.8	Cs-137
132	132	5.58	-5.2	Cs-137
133	133	5.93	0.7	Cs-137
211	211	5.80	-1.5	Cs-137, K-40
212	212	5.54	-5.9	Cs-137
213	213	6.14	4.3	Cs-137
221	221	5.77	-2.0	Cs-137, Mn-54
222	222	6.02	2.3	Cs-137, Mn-54
223	223	6.45	9.5	Cs-137, K-40, Co-57,
231	232	5.24	-11.0	Cs-137, K-40
232	232	5.37	-8.8	Cs-137
233	232	5.89	0.1	Cs-137
311	311	6.01	2.1	Cs-137, K-40, Ag-110m, Mn-54,
312	312	5.84	-0.8	Cs-137, K-40
313	322	6.22	5.7	Cs-137
321	321	6.04	2.6	Cs-137, K-40, Mn-54,
322	322	6.09	3.5	Cs-137, K-40
323	322	6.55	11.4	Cs-137, K-40, Cs-134, Mn-54,
331	332	5.34	-9.2	Cs-137, K-40, Co-57,
332	332	5.66	-3.8	Cs-137, K-40
333	332	5.97	1.4	Cs-137

The measurement results using the metal dummies are seen in Table 6~Table 7. The measurement error of Co-60 and Cs-137 was within -16% and -12% respectively. Thus, for the dummies in upper position (A~D), the measurement of the radioactivity was relatively accurate (within 6.1%). However, dummies in lower position (E~H), the measurements showed a relatively large error. The reason for this was because the location of the upper dummies was in "open" status, while for the lower dummies, they were blocked by 50mm of iron.

Table 6. Measurement result of metal dummy (Co-60)

Location	Sample Mass (kg)	Sample Height (cm)	Radioactivity (kBq)	Error (%)	Identification of nuclide
A	220	60	6.89	2.1	Co-60, K-40
B	220	60	6.83	1.1	Co-60, K-40
C	220	60	7.05	4.4	Co-60, K-40
D	220	60	6.97	3.2	Co-60, K-40
E	220	60	5.67	-16.0	Co-60
F	220	60	5.70	-15.5	Co-60
G	220	60	5.77	-14.5	Co-60, Mn-54
H	220	60	5.71	-15.4	Co-60, Mn-54

Table 7. Measurement result of metal dummy (Cs-137)

Location	Sample Mass (kg)	Sample Height (cm)	Radioactivity (kBq)	Error (%)	Identification of nuclide
A	220	60	6.15	4.6	Cs-137
B	220	60	6.16	4.6	Cs-137, K-40
C	220	60	6.25	6.1	Cs-137
D	220	60	6.15	4.5	Cs-137
E	220	60	5.47	-7.1	Cs-137
F	220	60	5.27	-10.4	Cs-137
G	220	60	5.41	-8.0	-
H	220	60	5.20	-11.7	-

The comparison test with the existing system in operation at power plants was done, as seen in figure 6, by attaching the elements between the dummies. The measurement was done in a total of 15 locations on both Co-60 and Cs-137,.

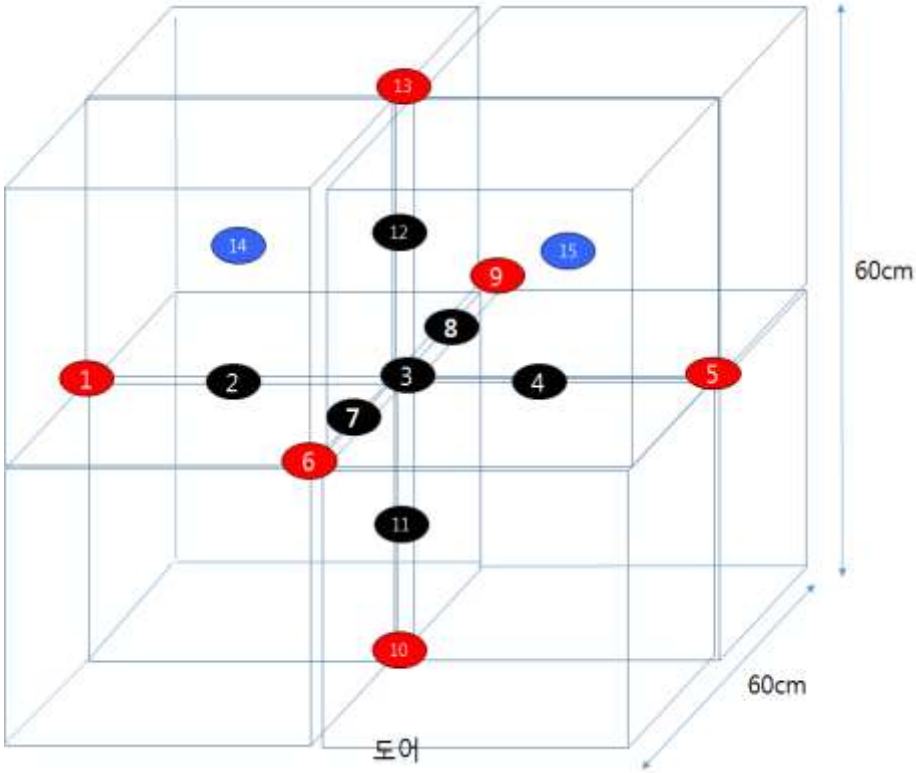

Figure 6. Location of element attachment (comparison experiment)

The measurement result is seen in figure 7 and 8. As shown in the figures, as the measured elements went inside the waste dummies, the error in comparison to the reference value relatively increased. The measurement was within 10% of the original radiation level, for the elements 1, 5, 6, 9, 10 and 13 that were attached to the outside of the dummy. However, the measurements in 2, 3, 4, 7, 8, 11 and 12 were carried out in locations where the elements were located between the iron and iron. The newly developed clearance monitor was evaluated to underestimate radioactivity by 40% whereas the current monitor in use by 95%.

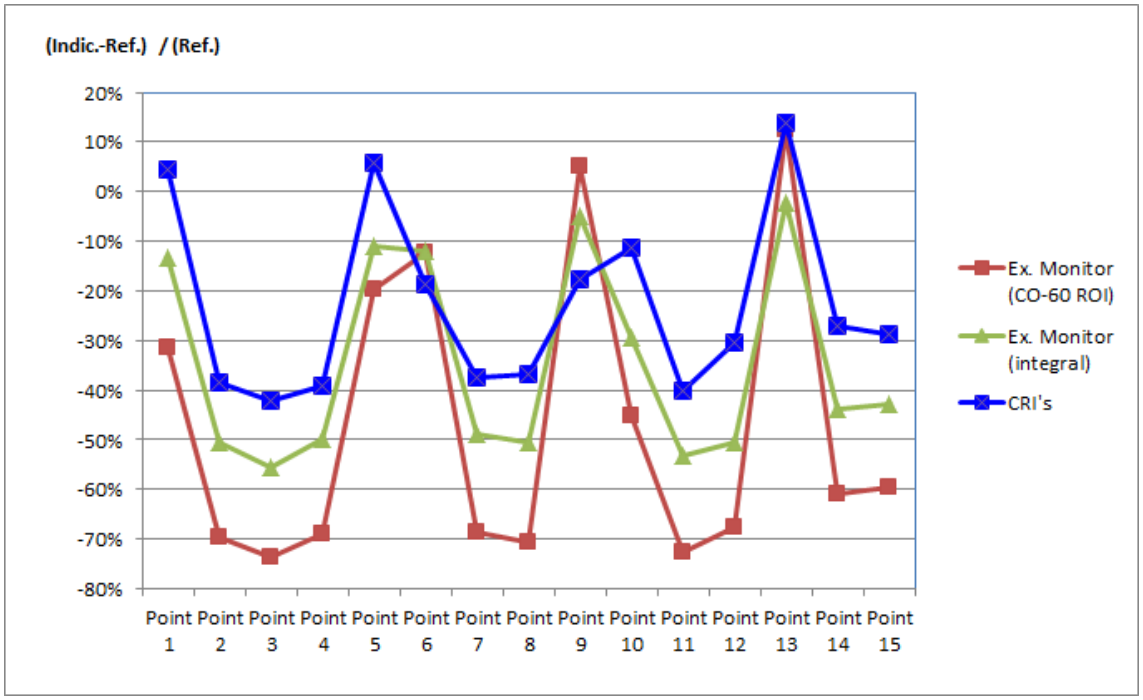

Figure 7. Result of comparison analysis (Co-60)

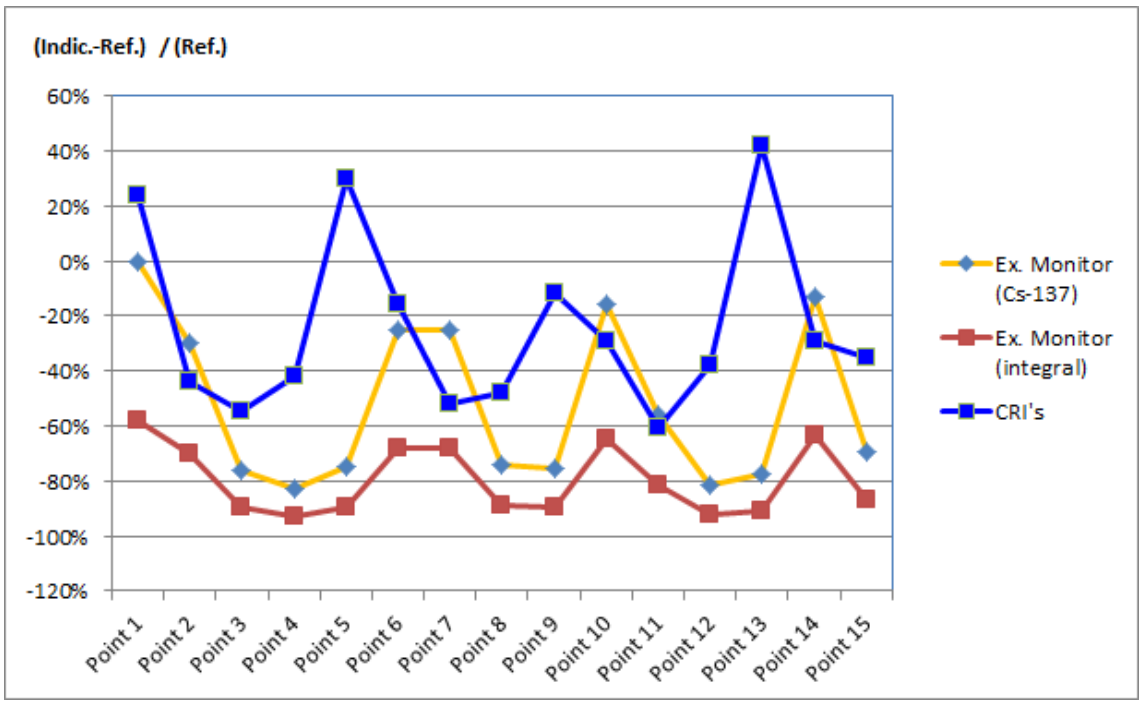

Figure 8. Result of comparison analysis (Cs-137)

Figure 9 shows a coordinate materialization function for users to effectively grasp the locations of the elements within the wastes.

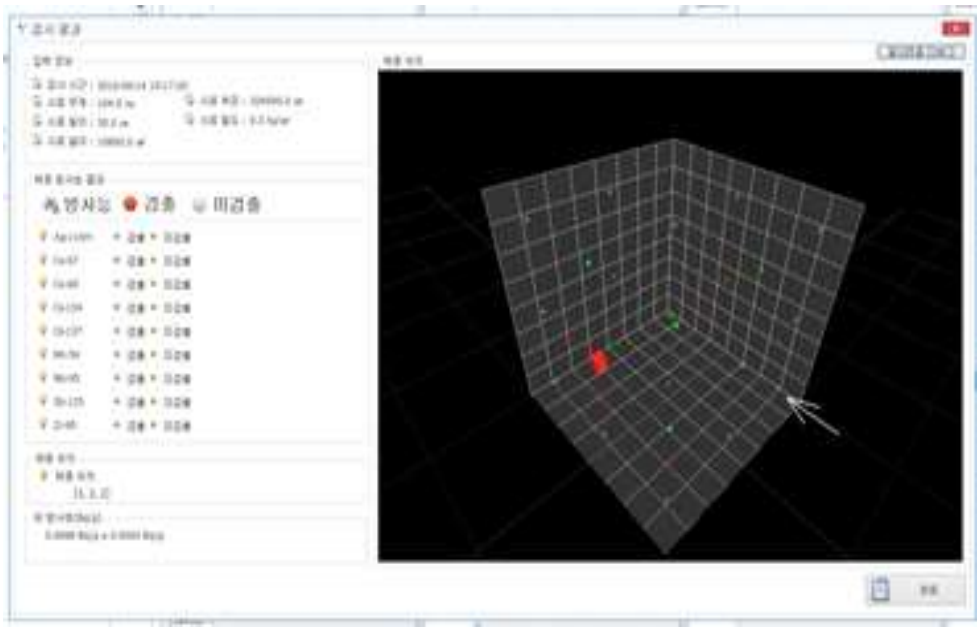

Figure 9. Function of identifying polluted location

CONCLUSION

The clearance monitor, developed by KHNP-CRI, increased the effective detection area by 71%, and the automatic withdrawal of the detector box from the system provides easy maintenance. Also, by making the measurement chamber a rectangle, specimen preparation became easier, and by making the shielding door open to the right and left, time of operation and stability were increased. Using the plastic scintillators and NaI scintillator, nuclide identification for radioactive wastes was made possible. In addition, the MDA influencing the performance of the clearance monitor was lowered to around 1/100 of the allowed clearance criteria. The channel shift phenomenon according to the temperature of NaI scintillator was automatically corrected using efficiency correction algorithm. Contamination location of radioactive waste, was realized by marking the 27 locations with graphics and coordinates.

Integrated operation and performance tests of CRI's clearance monitor were performed. The results of measurement applied to metal scraps (piles) embedded in standard sources (strengths of approximately 1/100 of the clearance criteria)

showed that errors were measured to be within -16% and -12% for Co-60 and Cs-137, respectively. The upper part of the pile showed relatively highly accurate measurement within 6%, while the lower part of the pile showed relatively larger errors. A comparison test with existing equipment was additionally conducted. As the radiation penetrated deeper into the piles, a relatively larger error was detected compared to the given standard value. In the case of measuring with the standard source attached on the outside of the pile, measurements were accurate within 10% of the standard value, while in the case of measuring with the standard source placed in the middle of the pile, the CRI's clearance monitor was undervalued in comparison to the standard value by a maximum of approximately 40%, and the existing equipment was undervalued by a maximum of approximately 95% for each channel and nuclide. Thus, it was verified that even under extreme conditions where the radiation source is in the middle of waste, a relatively more accurate measurement was possible in comparison to that obtained by existing equipment.

This year, the CRI's clearance monitor will be installed at one of the NPP site and tested using actual wastes and applicability other than metal scrap.

REFERENCES

[1] Development of Clearance Criteria of DAW, KHNP (TR), 2014

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[3] IAEA Safety Standards Series No. GSG-1 "Classification of Radioactive Waste', 2009