BALI 2013
SERIR
INTERNATIONAL CONFERENCE

ISBN : 978-602-70092-0-2

PROCEEDINGS

International Conference on the Sources, Effects and Risks of Ionizing Radiation

(SERIR2013)

October 10-11, 2013
Sanur Paradise Plaza Hotel
Bali, Indonesia

Published by
Center for Technology of Radiation Safety and Metrology
National Nuclear Energy Agency (BATAN)
March 2014

PROCEEDINGS

INTERNATIONAL CONFERENCE ON THE SOURCES, EFFECTS AND RISKS OF IONIZING RADIATION (SERIR)

BALI, 10-11 OCTOBER 2013

Organized by

Center for Technology of Radiation Safety and Metrology, National Nuclear Energy Agency

Supported by

UNSCEAR

March 2014

Organized and hosted by the

National Nuclear Energy Agency (BATAN)

In cooperation with

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR)

Supported by:

- Ministry of Research and Technology
- Ministry of Health
- Nuclear Energy Regulatory Agency
- Indonesian Radiation Oncology Society
- Indonesian Society of Radiology
- Indonesian Society of Nuclear Medicine

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Dr. Anhar R. Antariksawan, BATAN, Indonesia

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Members : Dr. Djarwani S. Soejoko, University of Indonesia, Indonesia

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FOREWORD

This publication represents the official record of the International Conference on the Sources, Effects and Risks of Ionizing Radiation held in Sanur, Bali, Indonesia, from 10 to 11 October 2013. This Conference was organized and hosted by the Indonesian National Nuclear Energy Agency (BATAN), in cooperation with the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), and supported by the Ministry of Research and Technology (Kemenristek), Ministry of Health (Kemenkes), Nuclear Energy Regulatory Agency (BAPETEN), Indonesian Radiation Oncology Society (PORI), Indonesian Society of Radiology (PDSRI), and Indonesian Society of Nuclear Medicine (PKNI).

The benefits and risks of any practice involving ionizing radiation, which is often shortened to just radiation, need to be established so that an informed judgment can be made on their use, and any risks minimized. The discovery of ionizing radiation and radioactive materials has led to advances in medical diagnosis and treatment, and they are used for a wide range of procedures in industry, agriculture, and research. Nevertheless, they can be harmful to living organisms, and they thus must be protected from unnecessary or excessive exposures.

The public, however, has a high sense of concern about radiation due to the accidents at nuclear power plants and other facilities, as well as by the common tendency to associate any form of radiation with all things 'nuclear', including nuclear weapons. The reasons for these concerns may be the lack of reliable and accessible information from the authorities and the scientific communities, and the misunderstandings that arise.

With this issue in mind, and considering current trends and developments in the use of ionizing radiation worldwide, the Conference aims to focus efforts in this area and to maximize the communication among the stakeholders (authorities, scientific communities and the public) so as to balance the knowledge on the benefits and risks of any practice involving ionizing radiation.

A number of about 200 participants attended this Conference, and consisted of researchers, hospital managers, medical professionals, government officials, decision makers and observers of those issues related with the topics of the Conference. Those participants came from both domestic and abroad.

The organization of the Conference was divided into three parts. The first one was plenary session presenting one keynote speaker and six invited speakers. Those speakers discussed the works of UNSCEAR and the current status of understanding and knowledge related to the levels and effects of ionizing radiation from several points of view. The second part was two panel discussions on the communication between scientific communities and the public on the issues of the levels and effects of ionizing radiation, and the cooperation among countries in gathering data related to those issues. The last part was presentations, both orally and in poster, of contributed papers of four topics of the Conference.

In total there were 38 papers submitted to the Conference, consisted of 33 from national participants and 5 from international participants. All papers were reviewed scientifically by international reviewer. Five of those papers are published in Atom Indonesia Journal, so that only their abstracts are communicated in these Proceedings.

This publication constitutes a record of the conference and includes: the opening and closing speeches, keynote address, slides that were submitted by invited speakers, all contributed papers, summaries of the discussion panels, and the President's summaries and conclusion.

***Disclaimer**: The material in these Proceedings has been edited by BATAN to the extent considered necessary to assist the reader. The views expressed remain, however, the responsibility of the named authors. The views are also not necessarily those of BATAN, the United Nations or the nominating organizations. Moreover, the designations employed and the presentation of material in this publication do not imply the expression of any opinion whatsoever on the part of the Secretariat of the United Nations concerning the legal status of any country, territory, city or area, or of its authorities, or concerning the delimitation of its frontiers or boundaries.*

WELCOMING ADDRESS
PRESIDENT OF THE CONFERENCE
INTERNATIONAL CONFERENCE ON THE SOURCES, EFFECTS AND RISKS OF IONIZING RADIATION

Sanur, Bali, 10 October 2013

His Excellency the State Minister of Science and Technology
Chairman of the National Nuclear Energy Agency
Secretary of the United Nations Scientific Committee on the Effects of Atomic Radiation,
Chairman of the Steering Committee,
Honorable Guests, Speakers and Participants

Assalamualaikum Warahmatullahi Wabarakatuh
Good Morning,

I am delighted to have the opportunity to welcome you to this importantInternational Conference on the Sources, Effects and Risks of Ionizing Radiation here in Sanur, Bali, Indonesia.It would be negligent of me if I did not first express my thanks to the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) for the cooperation to organize this Conference. I should also like to thank the Ministry of Research and Technology, the Ministry of Health, the Nuclear Energy Regulatory Agency (BAPETEN), the Indonesian Radiation Oncology Society (PORI), the Indonesian Society of Radiology (PDSRI) and the Indonesian Society of Nuclear Medicine (PKNI) for their support.

It is an honour, a pleasure and also a responsibility for me to be the President of this Conference. I will be at your service during these two days of Conference, so when you returnhome I hope you will feel that these coming two days are very valuable and hard to be forgotten.

Ladies and Gentlemen

As mentioned by the Chairman of BATAN, this Conference is held under an urgent need to give contribution to the works of the UNSCEAR. A Conference with the agenda tailored to the topic of discussion in the UNSCEAR is chosen as this has never been conducted anywhere before. The Conference is aimed to disseminate scientific findings related to the issues of sources, effects and risks of ionizing radiation.

To that end, the Organizing Committee invited contributions both academic and practice-based papers from member of the interested authorities, scientific communities and public on all aspects of the following topics:

- Exposures from Natural and/or Man-Made Radiation
- Occupational and Medical Radiation Exposures
- Health and Environmental Effects of Radiation
- Radiological/Nuclear Emergency Preparedness and Response

Honorable Guests, Speakers and Participants

We are delighted that despite his busy schedule, Dr. Malcolm Crick, the Secretary of the UNSCEAR, is willing to travel down 'half' under to this beautiful island of Bali, to be the keynote speaker at this Conference. There will also be six invited speakers – all of them are the prominent scientists in their own field, who will provide a comprehensive overview of the current status of the global sources, effects and risks of ionizing radiation.

This Conference also has an objective to increase awareness and improving understanding among stakeholders (scientific communities, regulatory authorities and general public) on the levels of ionizing radiation and its related health and environmental effects, as well as to enhance data collection on those issues.

Two panel discussions will deal with the above objectives. The first panel, deals with the communication among stakeholders, will consist of panelists representing scientific communities, international civil servant, government official, regulatory authority, and general public/media. This panel is expected to identify the ways and means of better communication on the discussed issues among the scientific community, regulatory authority and the general public.

The second panel will discuss the cooperation and collaboration among countries in the regional with the goal to gather more data on the level of radiation exposures and the effects of radiation to health and environment. We carefully asked some chairs of neighboring nuclear authority as panelist to discuss this particular issue, with the hope that the panel will come up with a commitment and action plan to work together to attain this goal.

Ladies and Gentlemen,

This Conference has attracted more than 100 participants from 6 countries. About 36 scientific papers will be presented by their authors orally or as posters. This event will offer you plenty of opportunities for extensive discussions, making of new contacts and strengthening the existing relationships after the oral presentations, during the poster sessions, while visiting the exhibition or at the other events.

In this afternoon, we invite all participants to a welcoming dinner held in this room. The world renowned Balinese traditional music and dance will be performed during the dinner. For participants who are interested, the Organizing Committee can arrange some sightseeing tours for the whole day on the next Saturday, 12 October 2013. Information regarding the tour is provided in the registration area.

Ladies and Gentlemen,

To enliven the Conference, the Organizing Committee has also invited some companies to display and exhibit their relevant products and technologies. I personally invite all participants to spare their time to look at the exhibition held at booths in the pre-function room.

In this occasion we would like to give appreciation to PT Murti Indah Sentosa& IBA Dosimetry which together has make a great contribution to the Organizing Committee as gold sponsor. The acknowledgement also goes to several other companies who make contribution to the successful of this Conference in one way or another.

Honorable Guests, Speakers and Participants

I look very much forward to this Conference, and I hope it is likewise forall participants. I hope there will be debates, because this Conference is open toeverybody and to all views, and certainly I will do my best to make

sure that thefloor is not monopolized by the invited speakers and panelists. So, you will haveyour chance to speak up, but please prepare your comments and questionscarefully.

Last but not least, we are sure that you will enjoy being in Bali, which has been crowned as “10 Great Dream Vacation 2013” by Huffington Post, and also as “World’s Most Romantic Islands 2013” by CNN Travel.

Thank you very much.

Wassalamu’alaikum warahmatullahi wabarakatuh.

President of the Conference
Dr. Anhar Riza Antariksawan

OPENING REMARKS
MINISTER OF RESEARCH AND TECHNOLOGY
INTERNATIONAL CONFERENCE ON THE SOURCES, EFFECTS AND RISKS OF IONIZING RADIATION
(delivered by Dr. Idwan Suhardi, Special Staff for Energy and Advanced Material)

Sanur, Bali, 10 October 2013

Chairman of the Steering Committee,
Secretary of the United Nations Scientific Committee on the Effects of Atomic Radiation,
Distinguished Guests, Speakers and Participants

Assalamualaikum Warahmatullahi Wabarakatuh
Good Morning,

First of all, we gratefully thank to Almighty Allah, for His kindness and blessing, so we all may gather this morning to attend the opening of the International Conferenceon the Sources, Effects and Risks of Ionizing Radiation, which is organized by the Indonesian National Nuclear Energy Agency (BATAN),in cooperation with the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), and supported by the Ministry of Research and Technology (Kemenristek), the Ministry of Health, the Nuclear Energy Regulatory Agency (BAPETEN), the Indonesian Radiation Oncology Society (PORI), the Indonesian Society of Radiology (PDSRI) and the Indonesian Society of Nuclear Medicine (PKNI).

I personally welcome the attendance of the prominent foreign scientists: Dr. Malcolm Crick, Dr. Abel Gonzalez, Dr. OhtsuraNiwa, Dr. Carl-Magnus Larsson, Dr. Peter Zagyvai, Dr. SompornChongkum, Dr. M. LebaiJuri and Dr. Ng Kwan Hoong. I also gratitude for the attendance of eminent local personalities: Prof. Dr. SoehartatiGondhowiardjo, Mr. MartuaSinaga, Mr. UsmanKansongand Dr. BambangBudyatmoko. Moreover, I do appreciating to all oral and poster presenters who have come along to this beautiful island of Bali and participate in this Conference.

Please allow me also to express my gratitude to all of you who have spent your valuable time to attend this special Conference. I would also like to extend my highest appreciation to the organizers for their hard works and efforts to organize this Conference.

Ladies and gentlemen,

Ionizing radiation and radioactive materials are widely used in medicine, industry, agriculture, environmental studies and research. These uses benefit each of us individually and the world's population as a whole.In Indonesia, the number of license for practices issued by the Nuclear Energy Regulatory Agency (BAPETEN) has steadily been arisen every year, indicating that these uses have also been more accepted by the Indonesians.

Besides bringinga huge benefit, the use ofionizing radiation and radioactive materials are also known to havedetrimental effects onhuman health and the environment. For this reason the radiation applicationsshouldberegulatedandclosely monitored so that those detrimental effects can be prevented to be manifest in human being and environment, and the risks of cancer and heritable effects, also known as stochastic effects, can be reduced to the extent reasonably achievable.

Distinguished guest, speakers and participants,

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) is the organization within the United Nations which has the mandate to undertake broad reviews of the sources of ionizing radiation and of the effects of that radiation on human health and the environment. The reviews and assessments perform by the Committee provide the scientific foundation used, inter alia, by the International Atomic Energy Agency(IAEA) in formulating international standards for protection of the public and of workers against ionizing radiation. The standards formulate by the IAEA are then used by its Member States as the basis to develop their own regulations, including by the Indonesian BAPETEN to regulate the radiation applications in Indonesia.

Indonesia is proud to be one of the only 27 countries in the world that are members of UNSCEAR. Indonesia became member of the UNSCEAR in 1973 by resolution 3154 C (XXVIII) of the UN General Assembly that decided to increase the Committee's membership and appointed Federal Republic of Germany, Indonesia, Peru, Poland and Sudan to be the member of the Committee.

To provide vision and direction for all its activities during the period 2009-2013, the Committee had developed a strategic plan. The Committee considered that its strategic objective for the period was to increase awareness and deepenunderstanding among authorities, the scientific community and civilsociety with regard to levels of ionizing radiation and the related healthand environmental effects as a sound basis for informed decision-making onradiation-related issues.

Ladies and gentlemen,

Since the beginning of its membership, Indonesia is committed to work closely with other members and the Committee's Secretariat for the successful of the Committee to fulfill its mandate. Indonesia actively participate in almost Committee's activities in Vienna, and provide some voluntary contributions to the trust fund to support the work of the Committee.

This Conference is also organized as a prove of Indonesia's commitment. The agenda of the Conference is drawn up by considering the Committee's strategic objective mentioned above. Through this Conference, therefore, Indonesia wishes toenhance its scientific contribution to the work of the Committee.

Honorable Guests, Colleagues, Ladies and Gentlemen,

I hope this conference could widen our knowledge on the issues of sources, effects and risks of ionizing radiation, balance its understanding and open a better communication among stakeholders on those issues. I also wish that the conference could formulate the coordination and collaborative works among countries in the regional, at least, to enhance the data collection related to the global levels of radiation exposures and its effects on health and environment.

Conclusively, through this opportunity, I declare officially the commencement of the International Conference on the Sources, Effects and Risks of Ionizing Radiation.

Thank you for all your patience and attention.

WassalamualaikumWarahmatullahiWabarakatuh

Minister of Research and Technology
Prof. Dr. Gusti Muhammad Hatta

SUMMARY AND CONCLUSIONS OF THE CONFERENCE

Report of the President of the Conference

Dr. AnharRizaAntariksawan
Deputy Chairman,
National Nuclear Energy Agency (BATAN)

BACKGROUND OF THE CONFERENCE

The International Conference on the Sources, Effects and Risks of Ionizing Radiation was held in Sanur, Bali, Indonesia, on 10-11 October 2013. The Conference was organized to enhance contribution from Indonesia to the works of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), as well as one of activities carried out in commemorating the 55th anniversary of the Indonesian National Nuclear Energy Agency (BATAN).

The organization of the Conference was divided into three parts. The first one was plenary session presenting one keynote speaker and six invited speakers, discussing the works of UNSCEAR and the current status of understanding and knowledge related to the levels and effects of ionizing radiation. The second part was two panel discussions on the communication between scientific communities and the public on the issues of the levels and effects of ionizing radiation, and the cooperation among countries in gathering data related to those issues. The last part was presentations of contributed papers of four topics of the Conference.

CONFERENCE OBJECTIVES

This Conference has the objectives to increase awareness and improving understanding among stakeholders (scientific communities, regulatory authorities and general public) on the levels of ionizing radiation and its related health and environmental effects, as well as to enhance data collection on those issues.

OPENING SESSION

Opening Session consisted of three opening addresses delivered by the current chair of the UNSCEAR, the Chairman of BATAN and the Expert Staff for Energy and Advanced Material of the Minister of Science and Technology, who was representing the Minister.

The current Chair of the UNSCEAR thanked Indonesia who had organized this International Conference, and also greatly welcome all participants in this Conference. The Chairman of BATAN highlighted the commitment of Indonesia to the works of the UNSCEAR, and the representative of the Minister, acted on behalf of the Minister, declared officially the commencement of the Conference.

PLENARY SESSIONS

Plenary sessions comprised of two parts:

(i) Keynote presentation

A keynote presentation was delivered by Dr. Malcolm Crick, Secretary of the UNSCEAR, who provided an overwiew of the work of the UNSCEAR. He underlined that the UNSCEAR is a pillar of the international radiation safety regime, whose mandate focusses on the scientific review of the levels and effects of radiation exposure.

(ii) Presentation by invited speakers

The current status of the global levels of radiation exposures was the main theme of Dr. Abel Gonzalez's speak in the first day. He explained the levels of radiation exposures from both natural and artificial radiations, as well as how they generates. Dr. OhtsuraNiwa of Fukushima Medical University then described in detail the Fukushima Daiichi nuclear plant accident and its consequences on radiation protection system and on the people.

In the second day, Dr. Peter Zagyvai described the emergency preparedness and response system developed by the IAEA. This followed by Dr. Carl Magnus Larsson who explained the the current knowledge of health and environmental effects radiation

Dr. SoehartatiGondhowiardjo, current chair of the Indonesian Radiation Oncology Society (PORI), described the current status of medical radiation exposures in Indonesia. The last speaker was Dr. SusiloWidodo of BATAN who described the works and results of BATAN's project in mapping of the environmental radiation and radioactivity levels in Indonesia.

PANEL DISCUSSIONS

During the two panel discussions, additional materials were presented and several important points were highlighted. This section captures some of the discussion from those panel discussions, which may have not been emphasized sufficiently elsewhere.

Panel Discussion 1: Increasing Awareness and Improving Underatanding among Stakeholders on the Levels of Ionizing Radiation and the Related Health and Environmental Effects

Some problems regarding these issues were discussed. Not only miscommunication between professional and the public that exist, professional often also feel unease with the media. Various terminology used by the professional are very confusing, and professional also tend to explain some safety issues scientifically yet difficult to comprehend, while the public simply need to know whether it is safe or not safe.

One of other issues difficult to be handled is probably the mistrust; scientific community do not trust the media, and the media in fact often do not trust the scientists. How to bridge the mistrust is therefore the question.

Risk communication is the best solution offered. Scientists must speak out and engage with the public. Scientists must learn from the public on how to communicate effectively. In this modern society that social media is very influential, scientists should also make use of this technology even more aggressive than just write a long piece of paper.

Panel Discussion 2: Coordination and Collaborative Works among Countries Toward Gathering Data on the Levels of Radiation Exposures and the Effects of Radiation to Health and Environment

The increasing use of radioactive sources and its complexity in many activities has been recognized by the panel. This might also increase the levels and effects of ionizing radiation globally.

To provide a better understanding on the levels and effects of ionizing radiation, the relevant data from various radiation applications should be available. Considering the variety of the data to be handled, coordination and collaboration among countries are seem inevitable.

To that end, countries in a region need to set up a way to collaborate by taking advantage the existing forum and networks or establish a new forum. This collaboration can then be used for data collection on the levels and

effects of radiation due to medical, public and occupational exposures. TENORM and consumer products are considered to be the priority for this collaboration in the region.

TOPICAL ISSUES

Topical issues consisted of contributing papers presented orally or in poster. In total there were 28 oral papers and 10 poster papers, which were grouped into four topical issues: (1) exposures from natural and/or man-made radiation, (2) occupational and medical radiation exposures, (3) health and environmental effects of radiation, and (4) radiological/nuclear emergency preparedness and response. All information contained in those contributing papers undoubtedly enrich the knowledge and understanding on the present levels and effects of ionizing radiation

CONCLUSIONS

Participants greatly appreciated the opportunity to attend this meeting, which focused on the sources, effects and risks of ionizing radiation. In general, scientific communities are dedicated to perform all the tasks needed of them in such a way that the levels of the sources, effects and risks of ionizing radiation are more to be known and understood. Nevertheless, collaboration among countries are needed to achieve that goals, and public at large should be provided with more access to understand better the issues. In both cases, many challenges in ways and means of collaboration and public communication remain to be solved.

It was admitted that this International Conference is the first of its kind to be organized, and the UNSCEAR expressed its satisfaction with its cooperation with BATAN in organizing it. It is therefore reasonable if the UNSCEAR could consider to organize the similar Conference in other regions to increase awareness and understanding on the levels and effects of ionizing radiation globally, to strengthen collaboration among countries in gathering data on those issues, as well as to introduce the UNSCEAR and its works to wider audiences.

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A. ***Exposures from Natural and/ or Man-Made Radiation***

B. ***Occupational and Medical Radiation Exposures***

KEYNOTE SPEAKER :

Dr. Malcolm Crick,

UNSCEAR

Austria

The works of the United Nations Scientific Committee on the Effects of Atomic Radiation

Malcolm Crick, Secretary, UNSCEAR
International Conference on the Sources, Effects and Risks of Ionizing Radiation, Bali, Indonesia, 10-11 October 2013

Contents

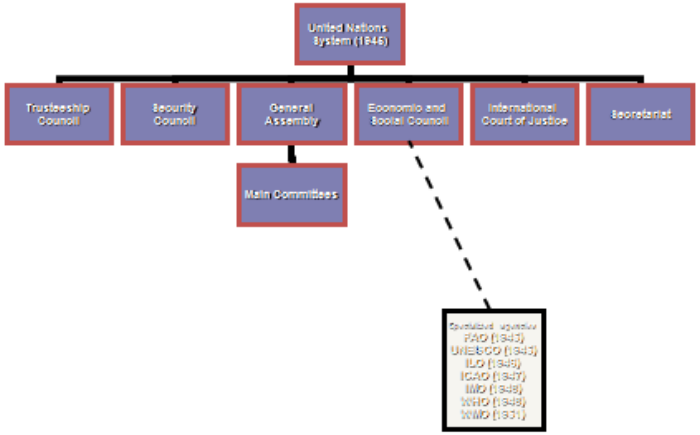
- Historical context
- Operations
- Recent studies and results
- Future outlook

Historical context

Charter of the United Nations (1945)

- *"We the peoples of the United Nations… United for a Better World"*
  - Maintain international peace and security
  - Develop friendly relations among nations…to strengthen universal peace
  - Achieve international cooperation in solving international problems of an economic, social, cultural or humanitarian character, promote human rights and freedoms
  - Centre for harmonizing actions of nations in achieving these common ends

Principal organs of the United Nations



The General Assembly

- First session opened 10 January 1946 at Central Hall, London, UK
- Chief, deliberative, policy-making and representative organ
- Comprises all 192 Member States
- Multilateral discussion of full spectrum of international issues in Charter
- Standard-setting and codification of international law
- Meets annually from September to December in New York

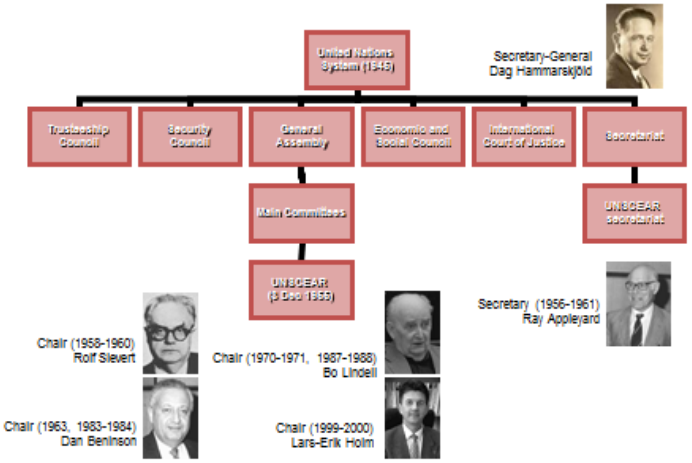
Nuclear weapons concerns

Hiroshima and Nagasaki bombings (1945)

Continuing weapons tests

Castle Bravo 1954. Bikini Atoll with fallout over inhabited islands and Japanese fishing vessel

Creation of Scientific Committee



Mandate

Review **levels, effects & risks of ionizing radiation**

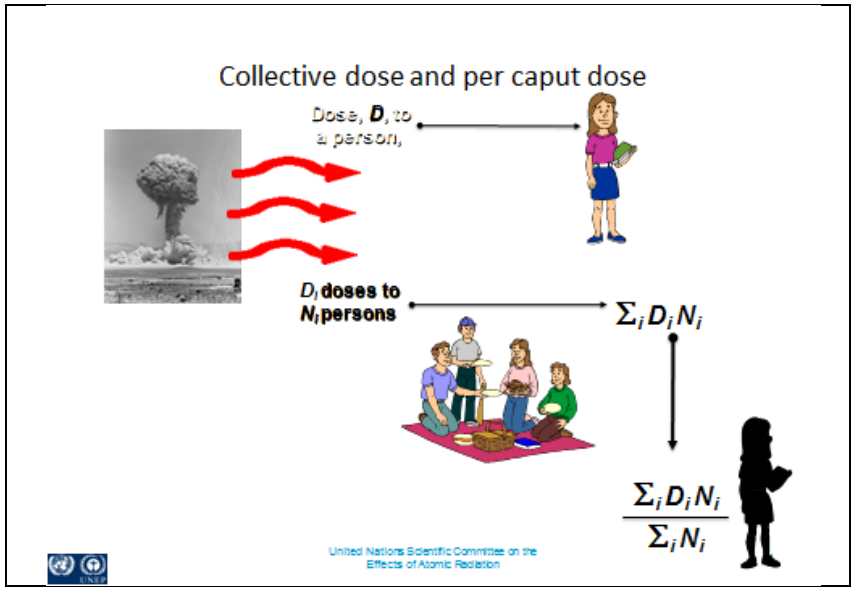
- Identify emerging issues
- Evaluate levels and effects
- Improve knowledge

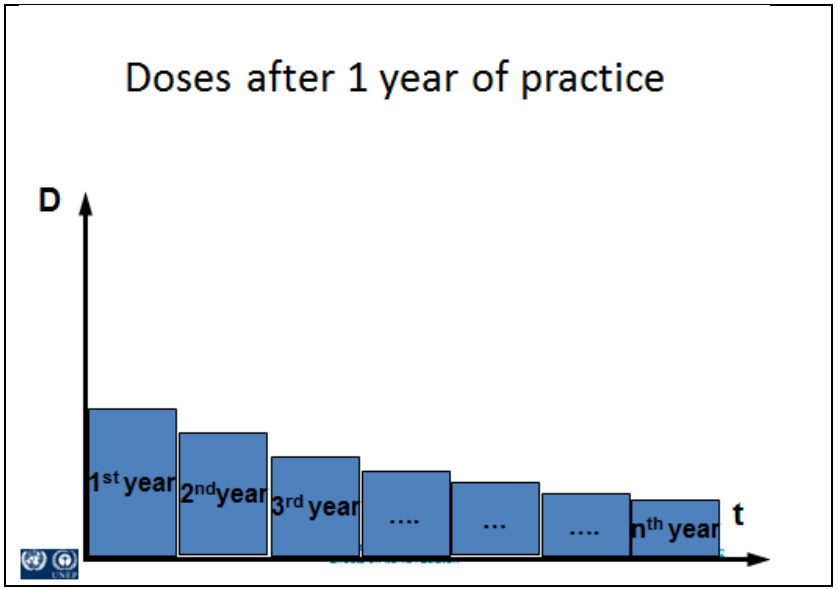
for General Assembly, scientific community & public

Recommend measurement standards
Identify research needs
Send team on request to country concerned about impact of nuclear weapons testing

Evaluation of the impact of nuclear weapons testing

- Collate and review reports from Governments
- Found major differences in measurement protocols
- Assess impact
- Needed new dosimetric concepts (e.g. collective doses and dose commitment)


Collective dose and per caput dose
Dose, D, to a person,
D_i doses to N_i persons
$\Sigma_i D_i N_i$
$\frac{\Sigma_i D_i N_i}{\Sigma_i N_i}$
United Nations Scientific Committee on the Effects of Atomic Radiation


Doses after 1 year of practice
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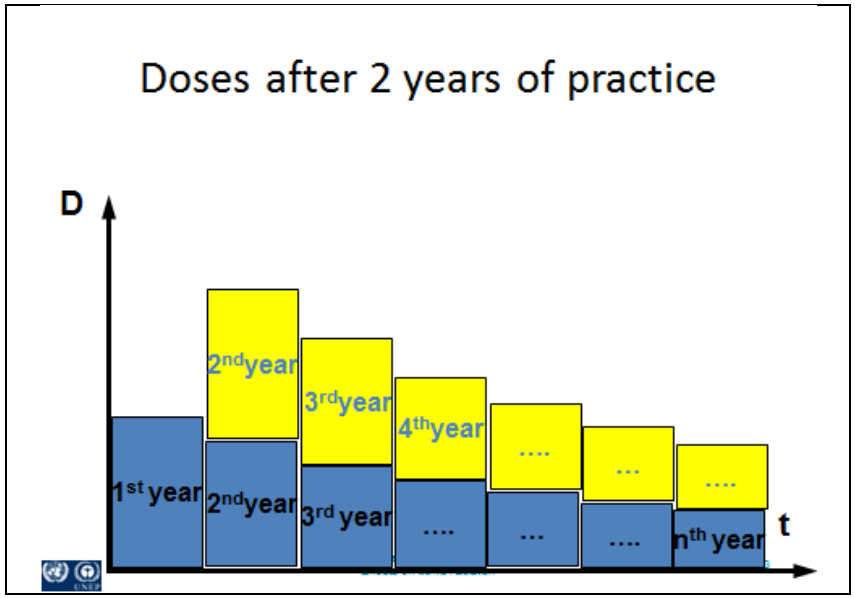
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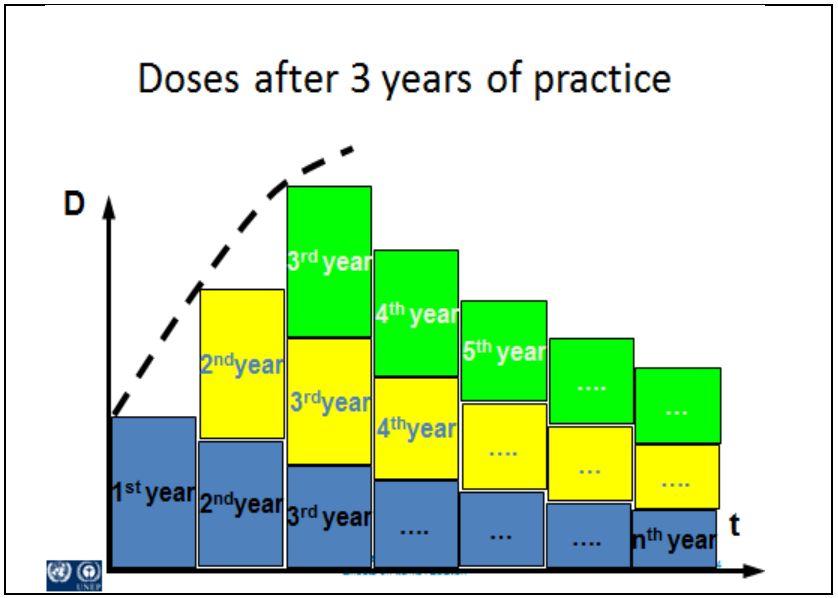
Doses after 3 years of practice
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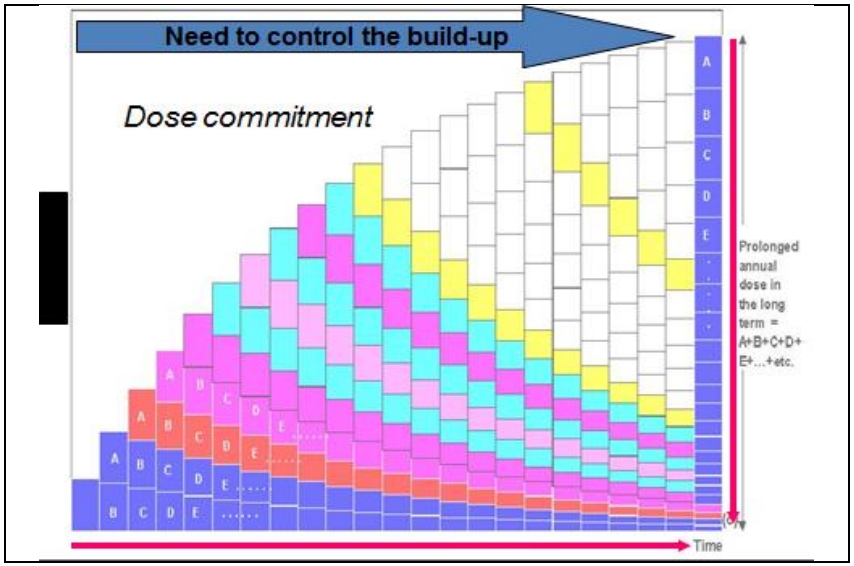


Estimated doses from different sources (UNSCEAR 1958 report)

Source	*per caput mean marrow dose for any 70-yr period (mSv)*
Natural	70
Occupational	1–2
Environmental	
Weapons tests cease end of 1958	1.6–9.6
Weapons tests continue for 100 years	28–170
Other man-made	70+

Hiroshima and Nagasaki

- Hiroshima
  - 70,000 immediate deaths
  - 90,000 to 140,000 by 1945
- Nagasaki
  - 40,000 to 75,000 immediate deaths
- 'Survivors'
  - 86,611 form primary source of information on radiation risks

Partial test ban treaty, 1963

Prohibited all test detonations of nuclear weapons in the atmosphere, outer space and underwater

Examples of technical underpinning

1963 partial test ban treaty

1972 UN Conference on Human Environment

UNEP

Recovery from Chernobyl accident

1996 reductions in international radiation exposure limits for workers and public

International action plans on worker, patient and environmental protection

2007 Recommendations of ICRP

States members of UNSCEAR

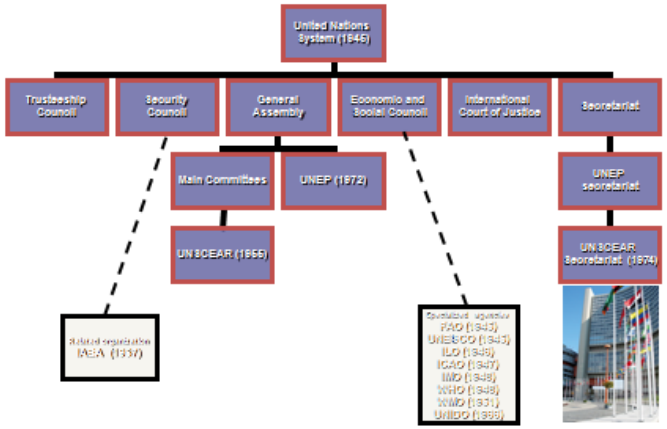
- Argentina
- Australia
- Belarus
- Belgium
- Brazil
- Canada
- China
- Egypt
- Finland
- France
- Germany
- India
- Indonesia
- Japan
- Mexico
- Pakistan
- Peru
- Poland
- Republic of Korea
- Russia
- Slovakia
- Spain
- Sudan
- Sweden
- Ukraine
- UK
- USA

1955
1973
1986
2011

Scientists from **27 UN Member States**

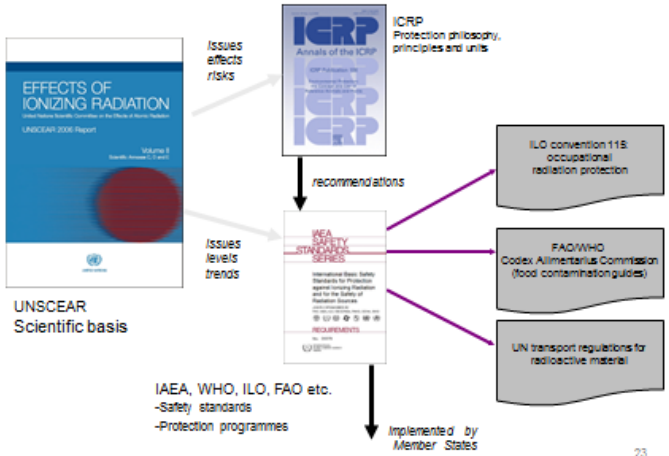
Other States & organizations provide relevant data

Secretariat provided through United Nations Environment Programme

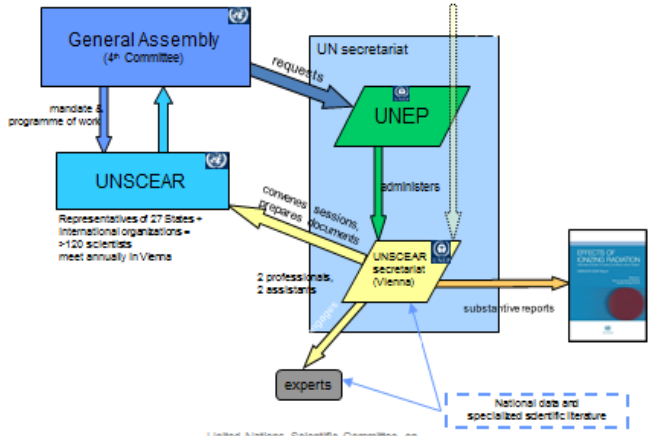


Operations

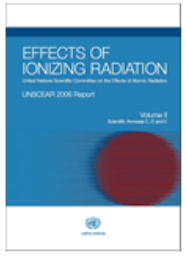
International radiation safety regime



Organizational context



Recent scientific findings



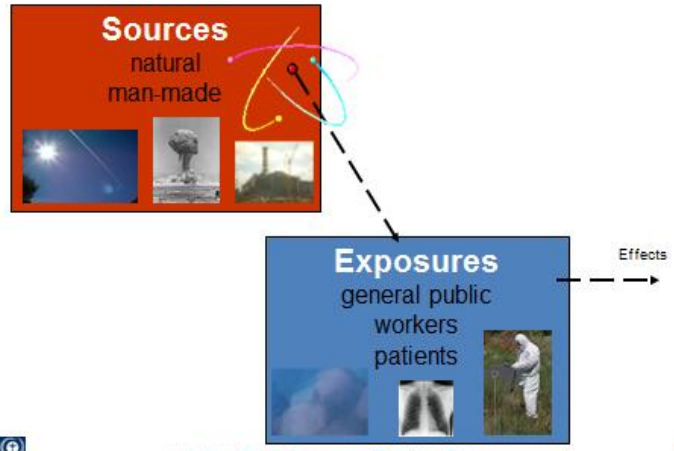
Last UNSCEAR reports

2006 2008 2010

2012 and 2013 in publication

Levels of exposure

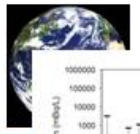
Reviews of levels

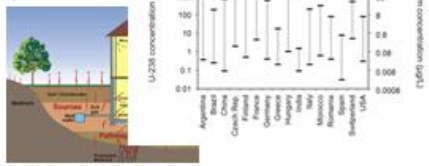


Natural sources

- Cosmic rays

- Terrestrial sources
  - Primordial radionuclides
    - Uranium-238 (4.5 billion years)
    - Potassium-40 (1.3 billion years)
    - Thorium-232 (14 billion years)
  - Exposures
    - External
    - Inhalation (radon)
    - Ingestion



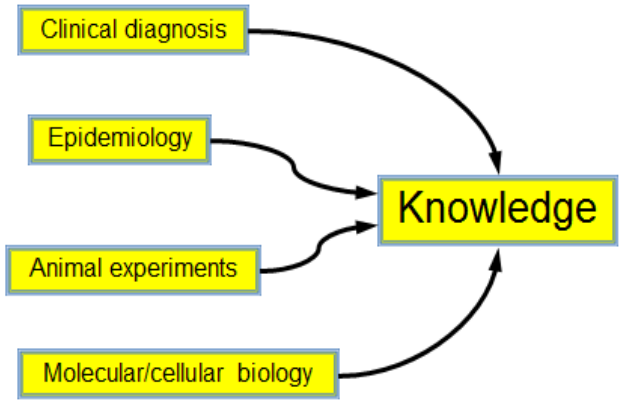


Artificial sources of exposure

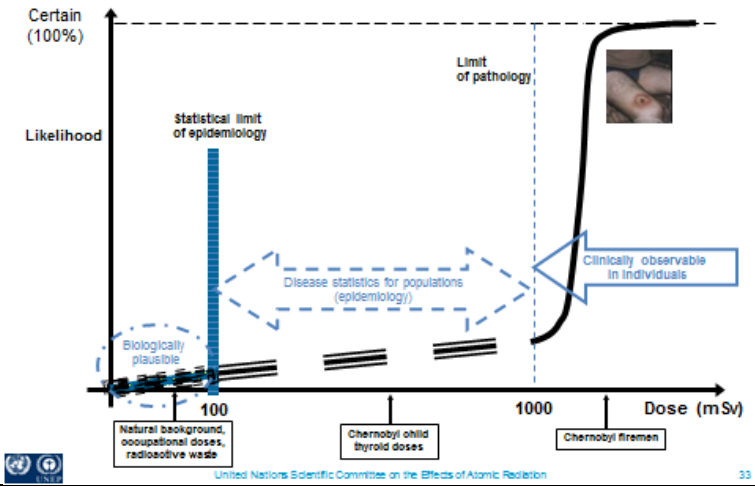
- Military activities
- Medical uses
- Occupational exposures
- Accidents
- Nuclear fuel cycle

Effects of exposure

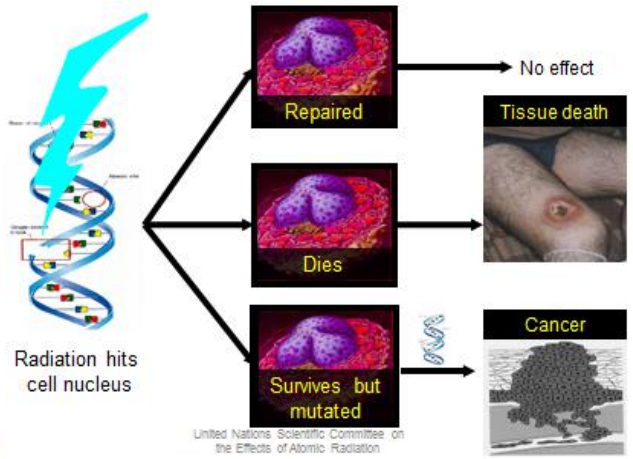
UNSCEAR's paths to knowledge



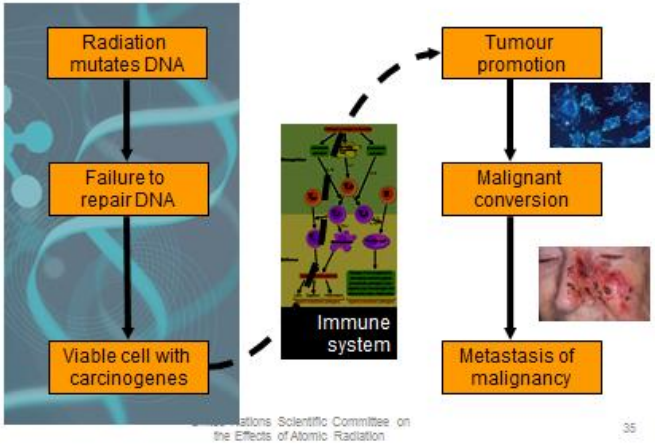
Radiation health effects



Radiation health effects



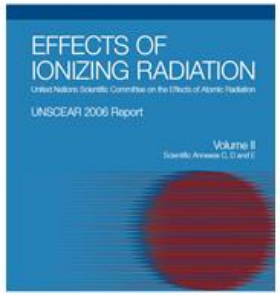
Prevalent opinion on radiation-induced cancer



2006 Report

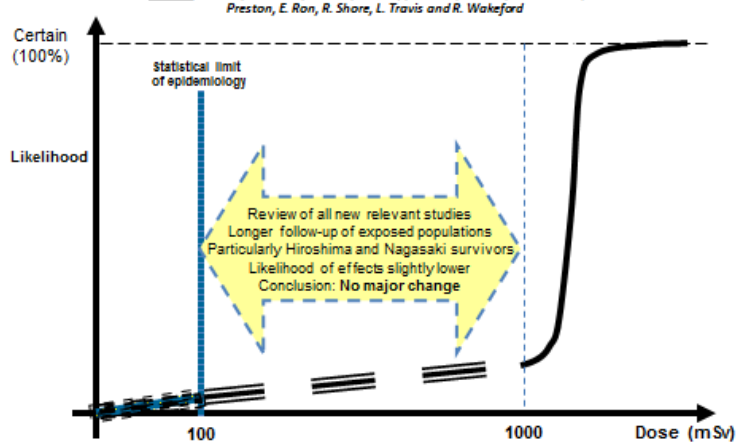
5 annexes in two volumes

- Cancer epidemiology
- Non-cancer epidemiology
- Non-targeted and delayed effects
- Immune system
- Risks of radon exposure

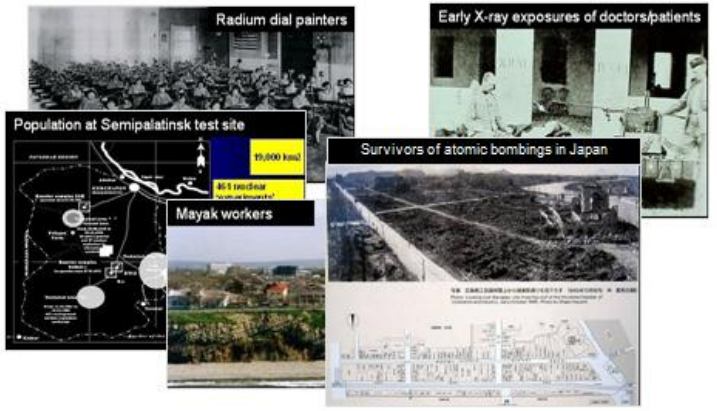


Annex A: Cancer epidemiology

M.Little, J. Boice, E. Gilbert, D. Hoel, P. Inskip, C. Land, J. Lubin, C. Muirhead, D. Preston, E. Ron, R. Shore, L. Travis and R. Wakeford

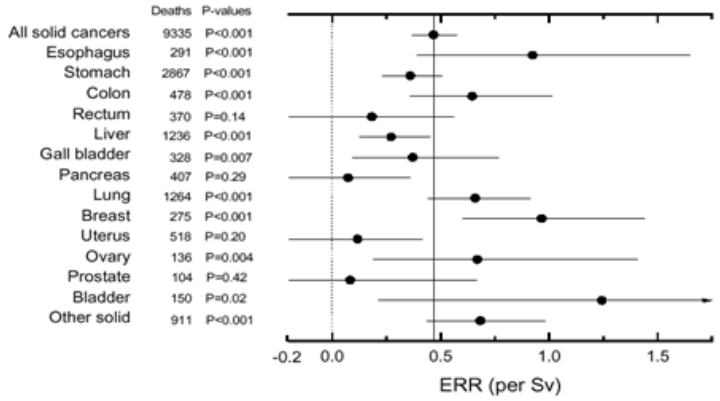


UNSCEAR reviewed many epidemiological studies



Site-specific solid cancer mortality

(ERR with 90% confidence intervals)



Lifetime risk estimates

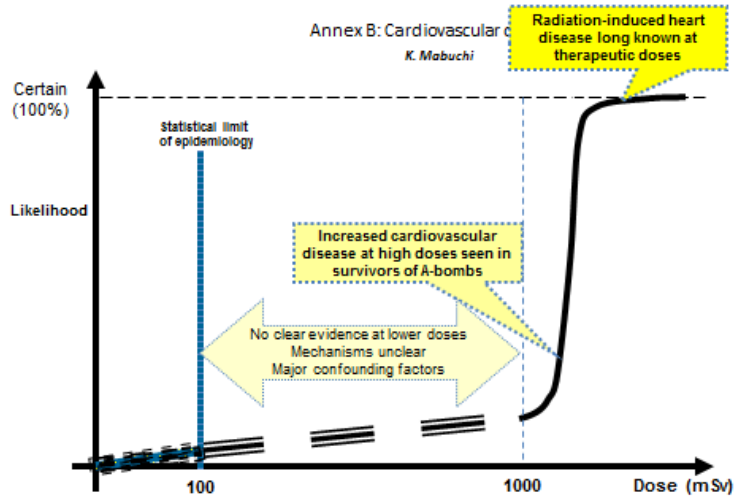
Averaging over five populations of all ages, both sexes

Risk of exposure-induced death (%/Sv)	*acute 0.1 Sv*	*acute 1 Sv*	*Previous estimate acute 1 Sv (DDREF=2)*
Solid cancer	3.6–7.7	4.3–7.2	11
Leukaemia	0.3–0.5	0.6–1.0	0.9

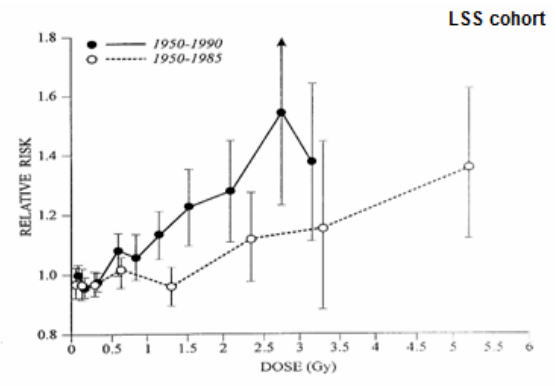
New dosimetry: 10% lower
Increased follow-up: 3–7% lower
Risk projection and transfer models: 35–40%
Uncertainties: up to 2 higher and include zero
Implicitly account for extrapolation to low doses (no need for DDREF)
Risks to children: 2–3 times higher

Annex B: Cardiovascular d

K. Mabuchi

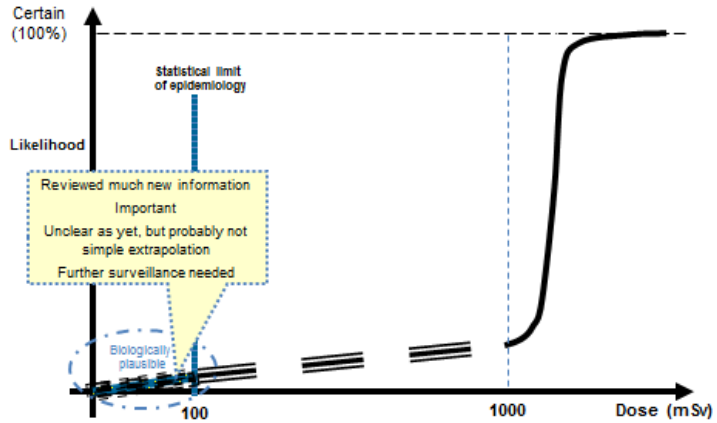


Radiation epidemiology of cardiovascular diseases



Annex C: Non-targeted effects

W. Morgan



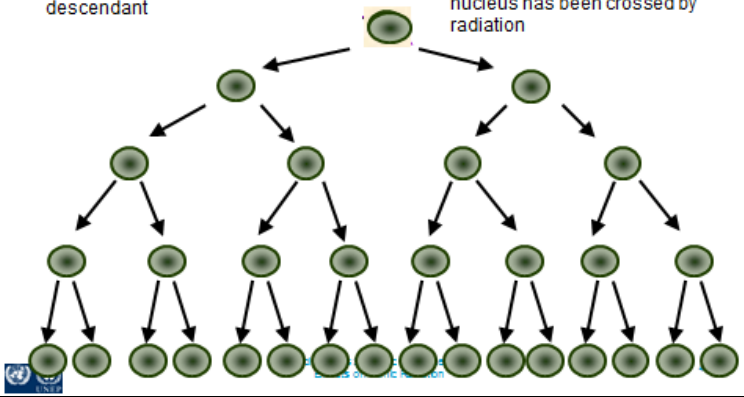
'Non-targeted' effects

- Genomic instability
- Bystander effects
- Abscopal effects
- Clastogenic plasma effects

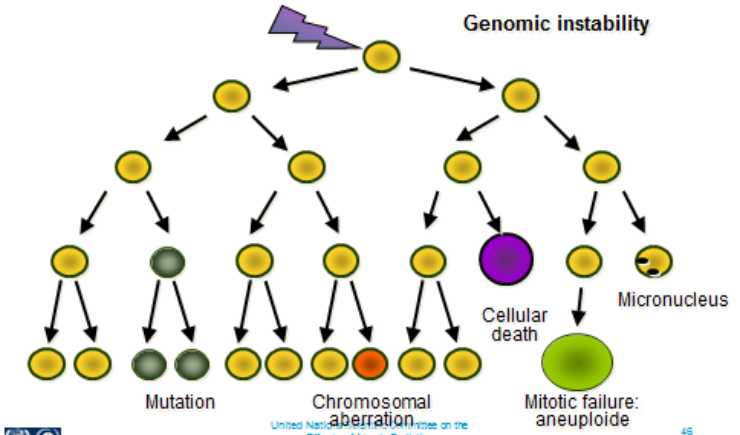
Basic paradigms of radiobiology

Damage fixed in DNA of irradiated cell, if not lethal, is transmitted to descendant

Effects occur in cells whose nucleus has been crossed by radiation

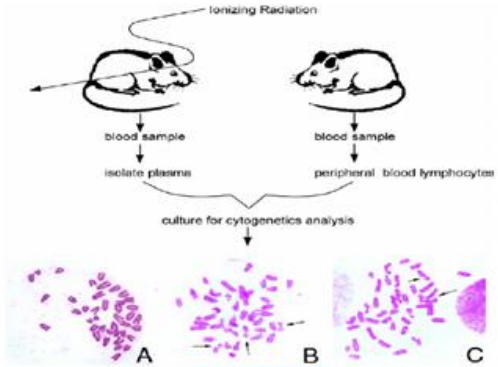


Challenge to the paradigm



Clastogenic plasma factors

- irradiated blood plasma seems capable of inducing chromosomal damage in unexposed cells



UNSCEAR conclusion on non-targeted and delayed effects

- Wealth of new information
- Instability and bystander effects well established
- Dominate at low doses and saturate with increasing dose
- Relevance to cancer and hereditary effects not clear
- Current risk estimates at moderate doses based on epidemiological observations encompass targeted, non-targeted and delayed effects
- Considerable debate regarding causal relationships between these effects and observed health effects

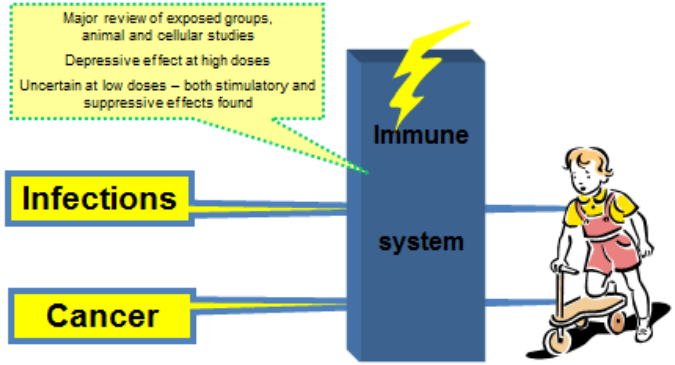
Annex D: Effects on the immune system

M.Perez, M.Bourguignon

Major review of exposed groups, animal and cellular studies

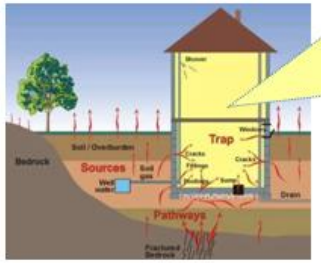
Depressive effect at high doses

Uncertain at low doses – both stimulatory and suppressive effects found



Annex E: Risks from radon

D.Chambers



Lung cancer is principal risk

New miner and residential data consistent

More work needed to account for time since exposure, attained age, dose rate and smoking

Given uncertainty, no change to UNSCEAR 2000 estimates

ERR at 100 Bq/m³ = 0.16
(90% of population risk occurs in smokers)

UNSCEAR will continue to convert radon exposure to effective dose using a convention of 9 nSv per Bq.h/m³

Overall conclusions of 2006 Report

- New wide-ranging review of health effects
- New cellular, genetic, microbiological studies
- Longer follow-up from epidemiology
- Differences at detailed level, but overall risk factors remain essentially unchanged

Effects on plants and animals (2008)

- Exposure levels at which population effects unlikely (1996)
- Reproduction more important than mortality
- New data from follow-up of Chernobyl
- **No change to overall 1996 conclusions**

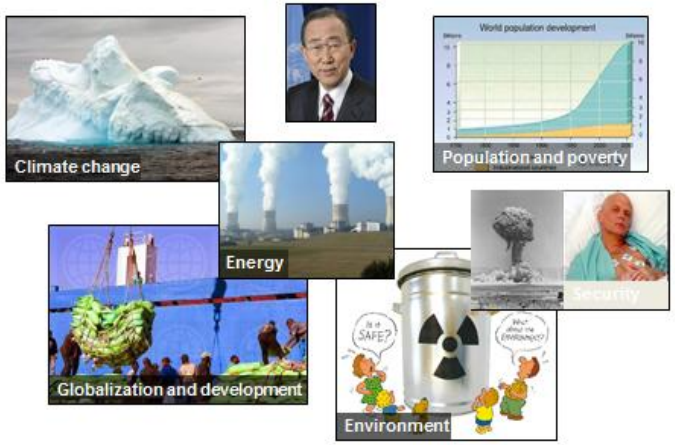
Overall conclusions on health risks

- Wide-ranging review
- New cellular, genetic, microbiological studies
- Longer follow-up from epidemiology
- Differences in details, but overall risks essentially from 2000 values

In the publication pipeline

- 2012 Report
  - Attributability of health effects to and inference of risk from radiation exposure
  - Uncertainty in cancer risk estimates for radiation exposure
- 2013 Report
  - Levels and effects of radiation exposure due to the accident at the Fukushima-Daiichi nuclear power station
  - Effects of radiation exposure on children

Global issues for UN system



Strategic plan 2014-2019

- Objective: increase awareness and deepen understanding among decision makers, the scientific community and civil society with regard to levels of exposure to ionizing radiation and the related health and environmental effects as a sound basis for informed decision-making on radiation-related issues.
- Major themes:

(a) global impact of energy production (including follow-up of 2011 accident at Fukushima Daiichi nuclear power station) and of rapidly expanding use of ionizing radiation in medical diagnosis and treatment; and

(b) radiation effects at low doses and low dose rates.

Operational shifts

(a) complete both wide-ranging summary reports on the levels and effects of radiation exposure and prepare special reports that respond to emerging issues as the need arises;

(b) use intersessional expert groups to develop assessment methodologies, conduct evaluations and maintain surveillance on emerging issues;

(c) develop networks of experts, scientific focal points in Member States and centres of excellence to facilitate access to expertise;

(d) enhance mechanisms for data collection, analysis and dissemination; and

(e) raise awareness and improve dissemination of the Committee's findings in readily understandable formats for decision makers and the public.

Concluding remarks

- **Fundamental** to international radiation **safety** regime
- Work affects decisions on energy debate, waste management, radiation medicine, protection of public, workers and environment; relevant in case of weapon deployment
- More efficient to develop global consensus through **sharing knowledge** than national or regional initiatives
- **Highly respected** by Governments, other international organizations and scientific community
- **Independence** and scientific objectivity
- Strategic plan to adjust to **future challenges**

PANEL DISCUSSION :

Dr. Peter Zagyvai,

IAEA

"International Conference on the Sources Effects and Risks of Ionizing Radiation", Bali Indonesia

Panel:

"Increasing Awareness and Improving Understanding among Stakeholders on the Levels of Ionizing Radiation and the Related Health

and Environmental Effects"

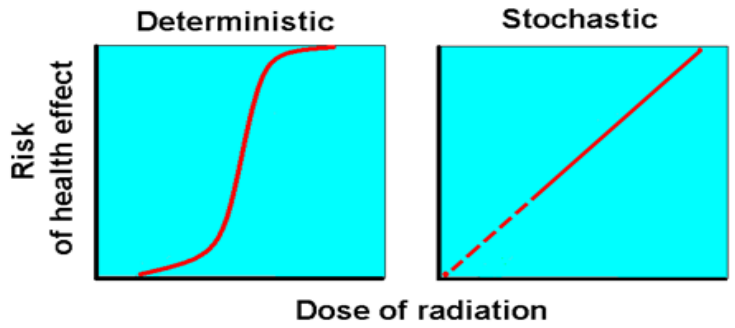
IAEA

Typical questions and concerns of the public if nuclear radiations are mentioned, particularly in a nuclear or radiological emergency

- Am I safe?
- Can children play outside?
- Should I take measures to decontaminate my home (such as remove all topsoil from my garden)?
- What does "radiation levels 20 times above normal" mean?
- What could be the consequences for my health?
- …

2

Effects of ionizing radiation



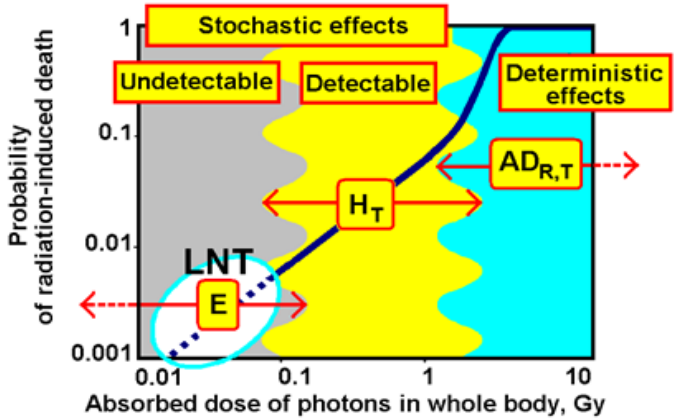
3

Plain language explanation is necessary

- These graphs are what we know.
- For better understanding it is also necessary to know that

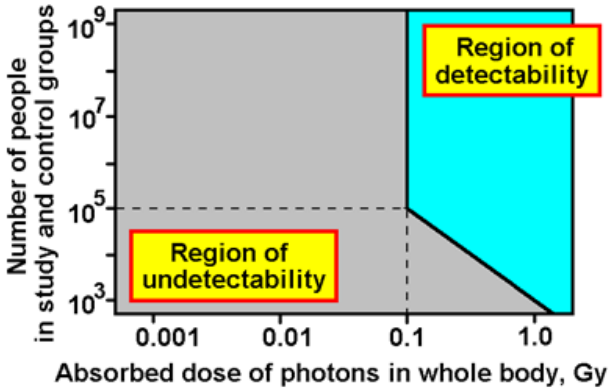
❑ what is the meaning of „risk";
❑ how the pertinent data were determined;
❑ how reliable these data are;
❑ how these data can be utilized.

Characterizing exposure situations



5

Statistical limitation of detectability



6

Lessons learned from past accidents

7

ABSC - Life Span Study mortality (1950-2002)

Diseases	Deaths			Attributable fraction
	observed	expected	excess	
Solid cancer	**6 718**	**6 205**	**513**	**8.3%**
Leukemia	**317**	**219**	**98**	**44.7%**

Database: 86 611 survivors with evaluated dose

Leukemia incidence started to increase 2 years after exposure; maximum increase occurred less than 10 years after exposure. Incidence of solid tumours started to increase at 5 years after exposure.

8

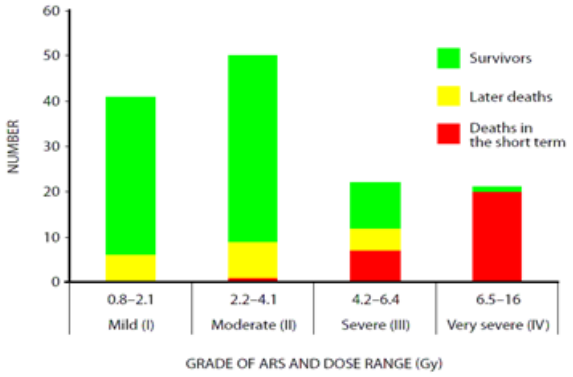
Deterministic effects after Chernobyl

Degree of Acute Radiation Syndrome	Range of RBE-weighted Whole body dose, Gy	Number of patients	Number of deaths
Mild (I)	0.8-2.1	41	-
Moderate (II)	2.2-4.1	50	1
Severe (III)	4.2-6.4	22	7
Very severe (IV)	6.5-16	21	20
Total	0.8-16	134	28*

* 28 persons died in 1986 from a combination of high external doses of γ-exposure and skin burns due to β-emitters

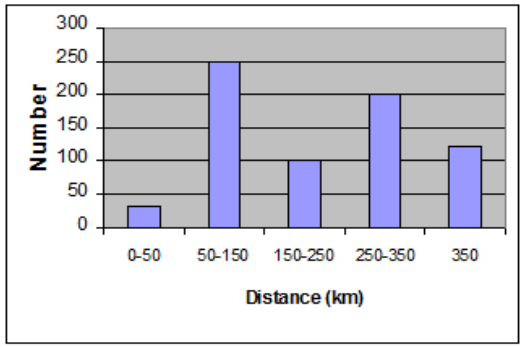
9

Outcome for Chernobyl NPP staff and first responders with ARS



10

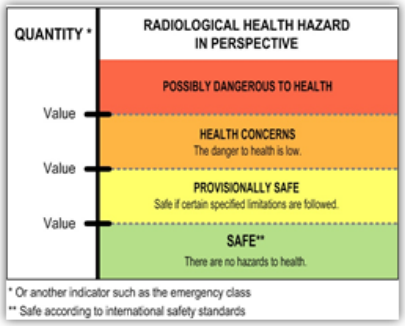
Total number of thyroid cancers in Belarus among those 0-18 years old at the time of Chernobyl accident



11

Communication with the public and decision makers

- Definition of 'safe' is required
- Measured operational quantities shall be interpreted in terms of the associated health hazard
- Safety = we can measure even what is insignificant



12

Psychological considerations

There will be better compliance with advice during an emergency if **trust** is maintained by:

- Clear and simple advice from experts
- Consistent advice and assessment (one official point)
- Use international guidance
- Maintain an ongoing information programme

13

THANK YOU

INVITED SPEAKER :

Dr. Peter Zagyvai,

IAEA

International Conference on the Sources Effects and Risks of Ionizing Radiation

Radiological / Nuclear Emergency Preparedness and Response

Peter Zagyvai
International Atomic Energy Agency
Centre for Energy Research, Budapest, Hungary

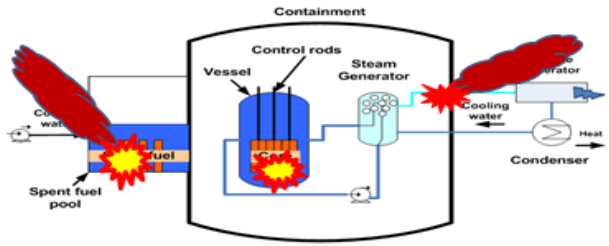
Emergency Preparedness and Response

1) Main objectives and elements of EPR
2) Summary of threat categories, planning distances and protective actions
3) Emergency classification for nuclear facilities
4) Generic criteria and practical guidance levels
5) Recent IAEA guidance for EPR

2

Off-site health effects are caused by major release

- Severe damage to fuel in the core of the reactor or spent fuel pool
- A major release (e.g. containment failure)



3

Main objectives of EPR activities

1) To regain control of the situation;
2) To prevent or mitigate consequences at the scene;
3) To prevent the occurrence of deterministic health effects in workers and the public;
4) To render first aid and to manage the treatment of radiation injuries;
5) To mitigate the risk of stochastic health effects in the population;
6) To prevent, to the extent practicable, the occurrence of non-radiological effects;
7) To protect, to the extent practicable, property and the environment;
8) To prepare, to the extent practicable, for the resumption of normal social and economic activity.

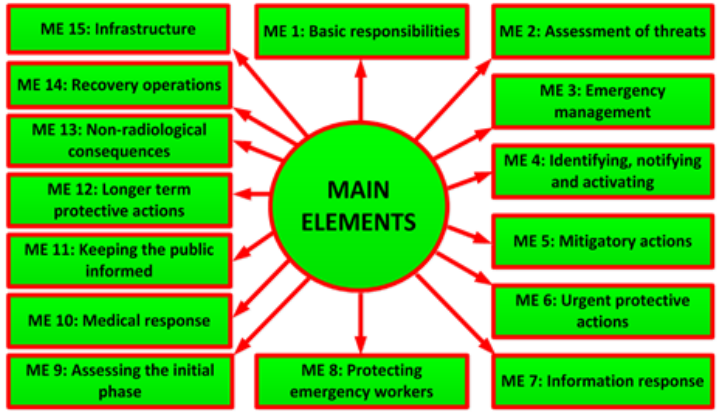
4

Goals of Emergency Preparedness

To ensure that arrangements are in place for a timely, managed, controlled, co-ordinated and effective response to any nuclear or radiological emergency at the scene and at the local, regional, national and international levels.

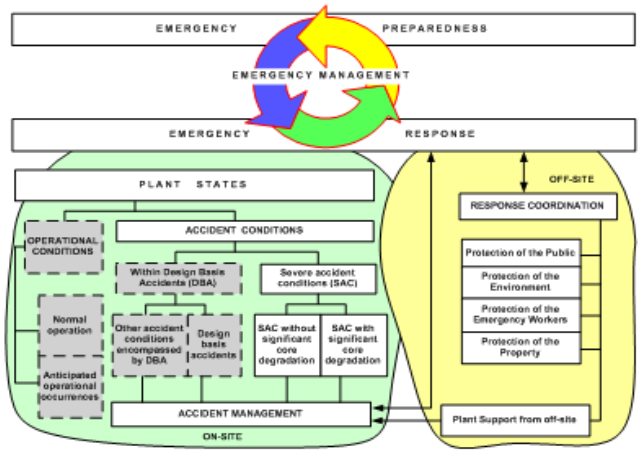
5

Main elements of EPR system



6

Emergency Management



7

Threat categories – facilities, activities

Threat category	Radiological threat
I	Reactor units of nuclear power plants with thermal capacity > 100 MWt (thermal) ≈ 30 MWe (electrical)
II	Smaller power reactor units, large research reactors, spent fuel storage facilities
III	Industrial irradiation facilities, small research reactors, LLW-ILW radioactive waste repositories
IV	Potential nuclear or radiological emergency in an unforeseeable location (e.g. lost source)
V	Effects of nuclear or radiological emergencies occurring in another country

8

Threat categories – possible consequences

Threat category	Radiological threat
I	Severe deterministic effects can be postulated on-site and off-site.
II	Warranting urgent protective actions off-site, deterministic effects are possible only on-site
III	No urgent protective actions off-site are warranted, severe deterministic effects only on-site
IV	Severe deterministic effects at an unpredictable location. Minimum level of capabilities for all States
V	Food contamination due to contamination caused by radioactive release necessitating food restrictions

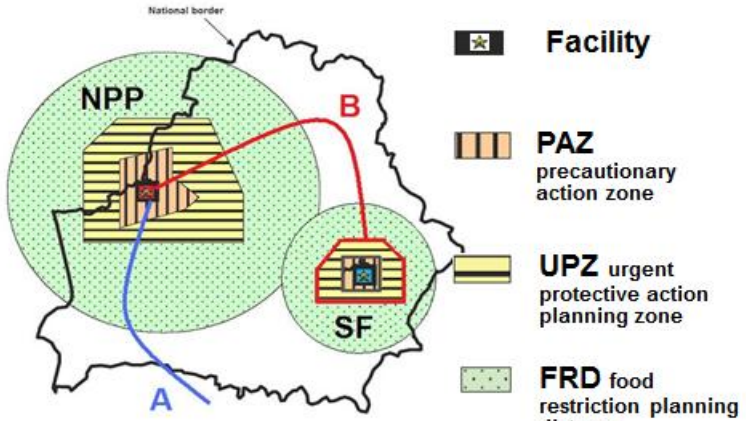
9

Protective actions in urgent, early and later phases of an emergency

- Iodine thyroid blocking (ITB)
- Evacuation
- Sheltering
- Relocation
- Prevention of inadvertent ingestion
- Decontamination of individuals
- Food, milk and water restrictions
- Identification and management of exposed people
- Protection of international trade and commercial interests
- Lifting/withdrawal of protective measures

10

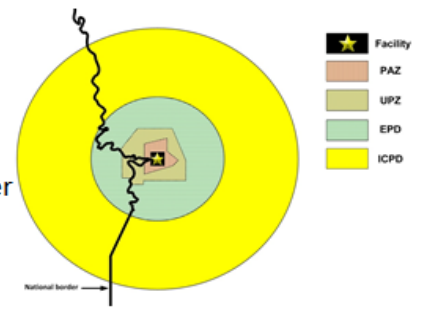
General schedule of planning zones



11

Planning zones update

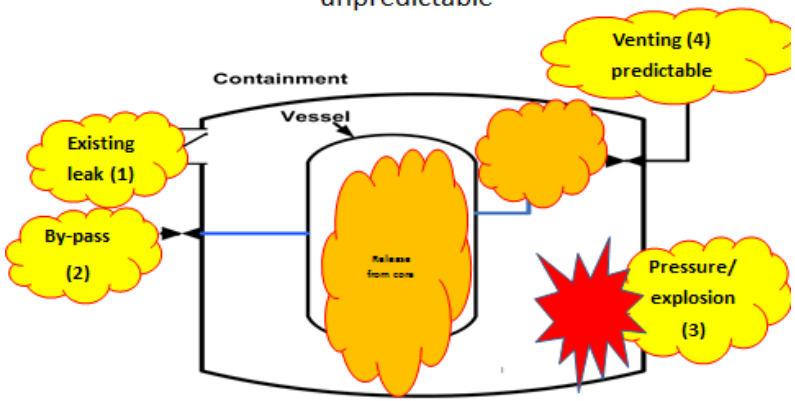
- Off-site ER organizations are responsible for implementing actions to protect the public
- Areas within 300 km of > 1000 MW_{th} nuclear power plant



PAZ: precautionary action zone UPZ: urgent protective actions zone
EPD: extended planning distance
ICPD: Ingestion and commodities planning distance

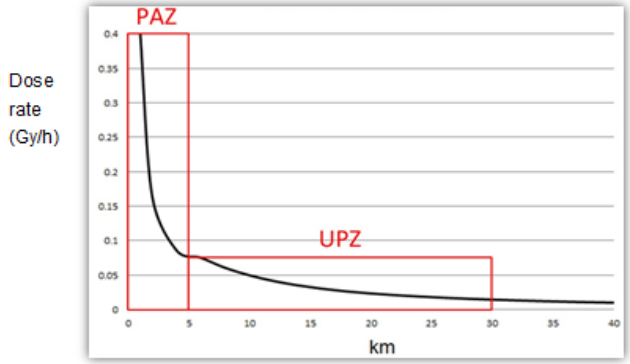
12

Vessel and containment failures are mostly unpredictable



13

Planning zones – projected doses in an assumed general emergency



14

Emergency planning zones: PAZ

A **precautionary action zone (PAZ)** is defined for facilities in threat category I. Arrangements shall be made here with the goal of taking precautionary urgent protective actions, before a release of radioactive material occurs or shortly after a release of radioactive material begins, on the basis of conditions at the facility (such as the emergency classification) in order to reduce, substantially, the risk of severe deterministic health effects.

15

Precautionary Action Zone

Precautionary actions taken before monitoring for severe emergencies:

- Prompt decision making (30 min)
- Promptly notify the public and recommend protective action (45 min)
- Iodine thyroid blocking (45 min – only possible if predistributed)
- Provisions for evacuation or sheltering the public (a.s.a.p. - no time guidance).

16

Emergency planning zones: UPZ and FRD

An **urgent protective action planning zone (UPZ),** for facilities in threat category I or II, for which arrangements shall be made for urgent protective action to be taken promptly, in order to avert doses off the site in accordance with international standards.

A **food restriction planning** distance **(FRD)** for facilities in threat category I or II is an area where agricultural produce would have to be restricted depending on the laboratory analysis of the food items.

17

Urgent protective action planning zone

- Prompt notification of the public to shelter and listen for additional information (1 h)
- Monitoring of environment and making decisions on additional protective action (4 hrs)
- Provisions for evacuation or sheltering following monitoring.

18

Food restriction planning distance

- Plans for monitoring agriculture, food and water
- Plans to take long-term protective actions:
  - Ingestion control
  - Relocation
  - Resettlement
- Similar actions are implemented in UPZ and PAZ.

19

Radii of emergency zones of nuclear reactors

Cat.	Power, MW_t	PAZ precautionary action zone	UPZ urgent protective action planning zone	FRD food restriction planning distance
I	> 1000	3-5 km	5-30 km	300 km
	100-1000	0.5-3 km	5-30 km	50-300 km
II	10-100	None	0.5-5 km	5-50 km
	2-10	None	0.5 km	2-5 km

20

Classification of emergencies

General emergencies at facilities in threat category I or II;

Site area emergencies at facilities in threat category I or II;

Facility emergencies at facilities in threat category I, II or III;

Alerts at facilities in threat category I, II or III;

Other emergencies at unpredictable locations associated with activities in threat category IV.

21

Emergency classification system for nuclear facilities

Excerpt from "EPR NPP Public Protective Actions" document


TABLE I. DESCRIPTION OF THE EMERGENCY CLASSES

Class	Description related to the protection of the public off the site	Example event resulting in this classification	Plain language explanation for the public
General Emergency	Events resulting in an actual or substantial risk of an atmospheric release requiring immediate implementation of urgent protective actions* and other response actions off the site within the PAZ, EPZ, EPD and ICPD.	• Actual or projected severe* damage of the fuel in the reactor core or the spent fuel pool. • Loss of safety functions projected to result in severe damage to the fuel in the reactor core or spent fuel pool, including loss of the ability to perform the following safety functions: – Shut the reactor down (establish reactor criticality control); – Keep the core covered (cool the fuel pool); – Remove decay heat from the reactor and the containment; – Maintain vital auxiliaries (e.g. AC/DC power and control systems, and instrumentation). • Detection of actual or imminent severe damage to the fuel in the reactor core or spent fuel pool. • Inability to control safety functions needed to protect the reactor core or spent fuel pool. • Detection of radiation levels off the site indicating actual severe damage to fuel (e.g more than 100 mSv/h).	An emergency has occurred at [insert name of the nuclear power plant], which means that a release of radioactive material is possible. If a release occurs, those located within an emergency zone or distance of the nuclear power plant may not be safe. You need to follow the instructions given by local officials that will provide for your protection, with priority being given to those who are most at risk. You need to immediately [insert appropriate instructions for the recommended protective actions and other response given in Table 4]. Local officials will communicate when the situation is safe and no protective or other response actions are required.
Site Area Emergency	Events resulting in a major decrease in the level of protection for those on the site and near the nuclear power plant requiring: (a) immediate alerting of the public to prepare to take protective actions and other response actions; (b) activation of the emergency response organizations; and (c) monitoring in the vicinity of the facility. Protective actions may need to be taken off the	• Conditions such that any additional failures could result in a General Emergency. • Potential to disrupt the performance of safety functions needed to protect the fuel in the spent fuel pool. • Releases from events* not involving severe damage to the fuel in the reactor core or spent fuel pool that warrant taking protective actions (e.g. ingestion restrictions based on monitoring and sampling off the site).	An emergency has occurred at [insert name of the nuclear power plant], which at this time does not represent a risk to your safety; however, it is prudent for you to prepare to [insert appropriate instructions for the recommended protective actions and other response given in Table 4] if it becomes necessary to ensure your safety. The following actions [insert action being taken, e.g. closing local schools] are being taken as precautionary measures. Local officials will communicate when the situation is safe and no protective or other response actions are required. We are now assessing the situation and you may need to take further actions once the assessment is completed.

Dose concepts as the basis of protective actions

Residual dose - The dose expected to be incurred after protective actions have been fully implemented (*or a decision has been taken not to implement any protective actions*).

Projected dose - The dose that would be expected to be received in the absence of (*some*) planned protective actions.

Received dose - The dose that is incurred after protective actions have been fully implemented (*or a decision has been taken not to implement any protective actions*).

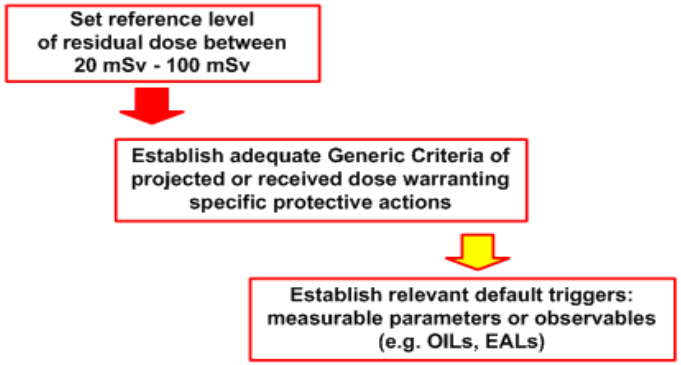
23

Basic Principles of Protective Strategy

- All possible efforts should be made to prevent severe deterministic health effects and to reduce the occurrence of stochastic effects.
- A reference level expressed in terms of residual/projected dose shall be set, typically an effective dose in the range of between 20 and 100 mSv that includes dose contributions via all exposure pathways.
- The protection strategy shall include planning for residual doses to be as low as reasonably achievable below the reference level, and the strategy shall be optimized.

24

Development of Protective Strategy



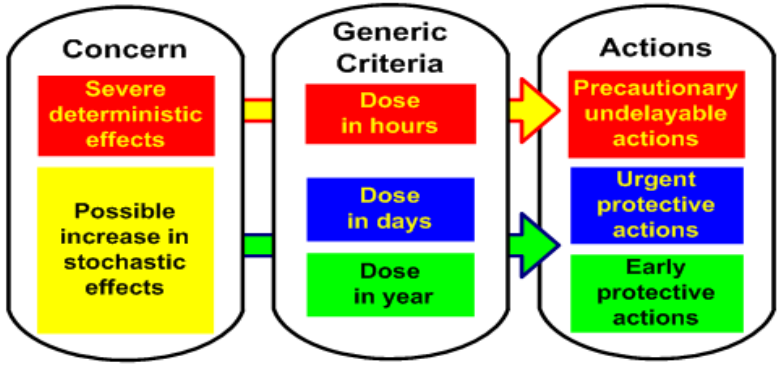
25

Basic Principles of Generic Criteria

- On the basis of the outcome of the optimization of the protection strategy, using the generic criteria for particular protective actions and other actions, expressed in terms of projected dose or dose that has been received, shall be developed. If the numerical values of the generic criteria are exceeded, those protective actions and other actions, either individually or in combination, shall be implemented.

26

Generic Criteria for Protective Actions



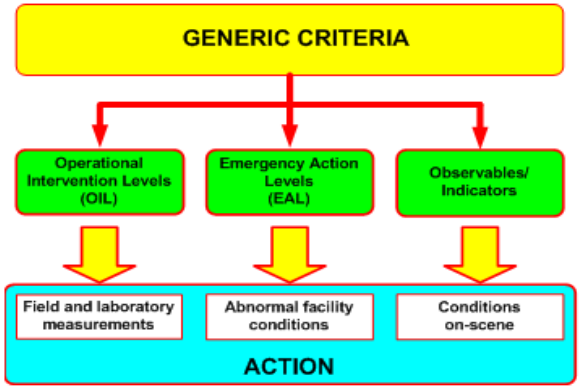
27

Basic Principles of Operational Triggers

- Once the protection strategy has been optimized and a set of generic criteria has been developed, pre-established default triggers for initiating the different parts of an emergency plan, primarily for the initial phase, shall be derived from the generic criteria. Default triggers, such as observable on-scene conditions, emergency action levels (EALs) and operational intervention levels (OILs), shall be expressed in terms of parameters or observable conditions.

28

Use of Generic Criteria



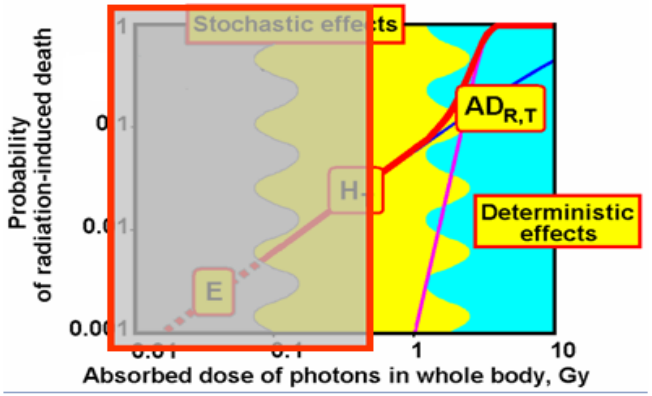
29

System of Response Actions

Possible health effects of exposure	Basis for implementation of protective and other response actions	
	Projected dose	Received dose
Severe deterministic health effects	Implementation of precautionary urgent protective actions, even under adverse conditions, to prevent severe deterministic health effects	Other response actions for treatment and management of severe deterministic health effects

30

Evaluation of High Dose Exposure



Generic Criteria for Controlling Severe Deterministic Health Effects

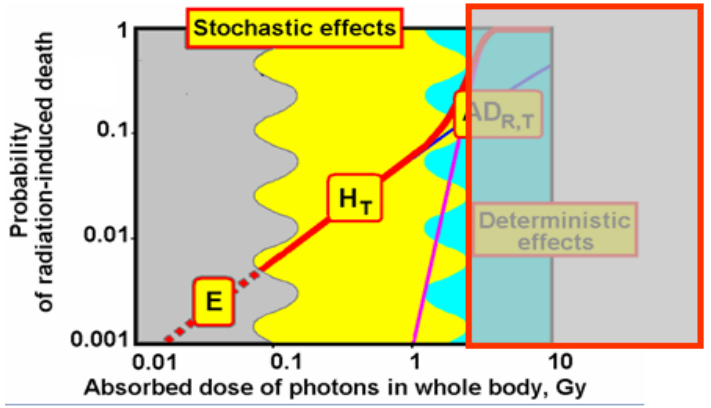
Acute external, local and contact exposure

Organ or tissue	RBE-weighted dose
Red marrow	1 Gy
Foetus	0.1 Gy
Soft tissue	25 Gy at 0.5 cm to 100 cm^2
Skin derma	10 Gy at 0.4 mm to 100 cm^2

System of Response Actions (2)

Possible health effects of exposure	Basis for implementation of protective and other response actions	
	Projected dose	Received dose
Increase in stochastic health effects	Implementation of urgent protective actions and initiation of early protective actions to reduce the risk of stochastic health effects as far as reasonably possible	Other response actions for early detection and effective management of stochastic health effects

Evaluation of Low Dose Exposure



Generic Criteria for Controlling Stochastic Health Effects (1)

Dosimetric quantity	Projected dose in 7 days and protective actions	
Total effective dose	**100 mSv**	Sheltering, evacuation, decontamination, restriction of food consumption …
Total equivalent dose in foetus or embryo	**100 mSv**	
Committed equivalent dose in thyroid	**50 mSv**	Iodine thyroid blocking

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Generic Criteria for Controlling Stochastic Health Effects (2)

Dosimetric quantity	Projected dose in a year and protective actions	
Total effective dose	**100 mSv**	Temporary relocation, decontamination, replacement of food, milk and water, public reassurance …
Total equivalent dose in foetus or embryo (for period of *in utero* development)	**100 mSv**	

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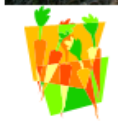
Basic Principles of Operational Triggers

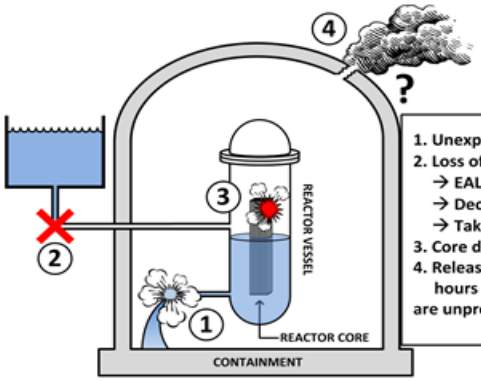
International guidance specifies:

- "Operational intervention levels" (OILs) – calculated quantities that correspond to one of the generic criteria and can be measured directly
- "Emergency action levels" (EALs) – specific, predetermined, observable operational criteria used to detect, recognize and determine the emergency class of an event at facilities in threat categories I, II and III.



37

EALs provide early on-site warning



1. Unexpected event
2. Loss of safety function (SF)
   → EAL exceeded
   → Declaration of GE
   → Take urgent actions
3. Core damage
4. Release to the atmosphere in hours or days (time, duration and size are unpredictable)

38

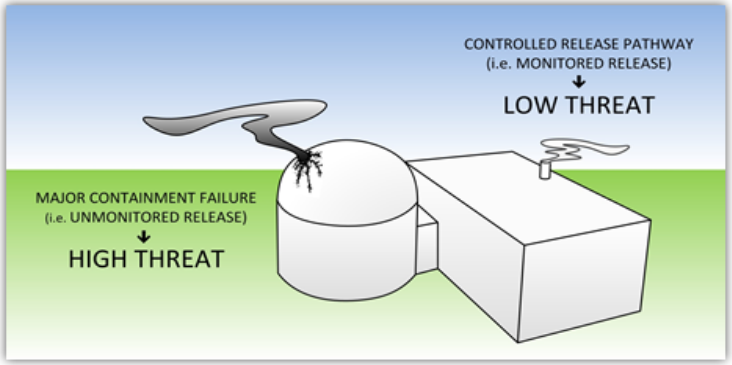
Example EALs for General Emergency – (projected or actual severe damage to fuel)

Before severe fuel damage (loss of SF)	Imminent severe fuel damage	Actual severe fuel damage	After release, monitoring off-site
Projected loss of AC and DC for a site-specific time (e.g. 40 min)	Core temperature (CET) >800°C	In-plant radiation levels increase	Dose rates >100 µSv/h

Classify as soon as you can

39

Off-site threat: unmonitored pathway



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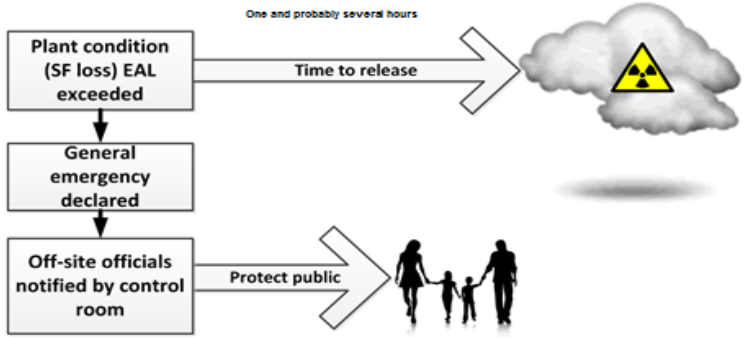
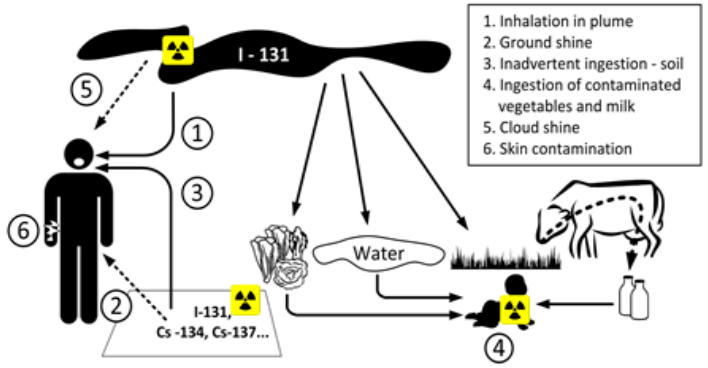
Goal: to protect the public before a release



41

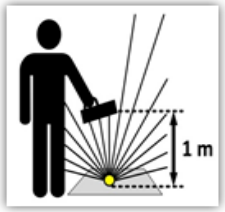
Exposure pathways – targets of OILs

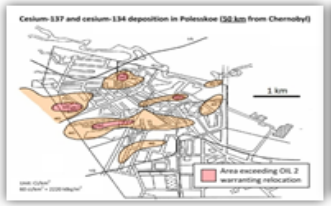


42

Operational intervention levels – monitoring and evaluation

- Operational intervention levels (OILs) – default or specific values
- Plain language explanations for OILs
- Contamination and hotspots
- Displaying monitoring results





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Operational Intervention Levels

International guidance specifies the following OILs:

- OIL1 is a measured value of ground contamination calling urgent protective and medical actions to avoid 100 mSv in 7 days
- OIL2 is a measured value of ground contamination calling for early protective actions to keep the dose to any person living in the area for a year below the generic criteria
- OIL3 is a measured value of ground contamination calling for immediate restrictions of consumption of leaf vegetables, milk from animals grazing in the area and rainwater to keep the dose to any person below the generic criteria for taking urgent protective action to avoid 10 mSv in a year.

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Operational Intervention Levels (cont.)

International guidance also specifies the following OILs:

- OIL4 is a measured value of skin contamination calling for performing decontamination or providing instructions for self-decontamination and for limiting inadvertent ingestion and to initiate medical treatment or screening, as required.
- OIL7 is a measured value of concentrations of indicator radionuclides ^{131}I and ^{137}Cs in food, milk or water that warrant the consideration of restrictions of consumption so as to keep the effective dose to any person below 10 mSv per annum.
- OIL8 is a measured value of dose rate from the thyroid gland caused by radioiodine uptake of the examined person.

45

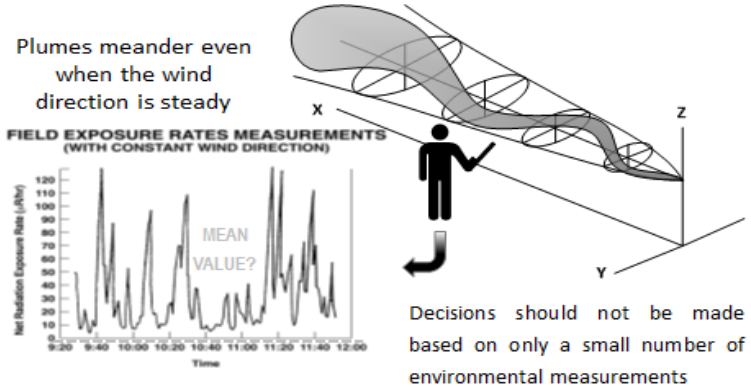
Measurement of OIL dose rate

Quantity	Symbol	Unit	Monitoring
Personal dose equivalent	$H_p(10)$	Sv	External exposure of the individual (dosimeters)
Ambient dose equivalent	$H^*(10)$	Sv	Environment and public monitoring of external radiation field (dose rate)

46

Large variations of monitoring results are expected

Plumes meander even when the wind direction is steady



Decisions should not be made based on only a small number of environmental measurements

47

Basis of OILs – relation to dose consequence

Inhalation of ^{131}I

100 mSv *effective* dose → Misleading because of the more significant health hazard

2000 mSv *equivalent* dose to thyroid → Possible severe effects in thyroid

48

Emergency Workers

An emergency worker is any person having a specified role as a worker in an emergency and who might be exposed while taking actions in response to the emergency. Emergency workers may include those employed by registrants and licensees as well as personnel from response organizations, such as police officers, firefighters, medical personnel, and drivers and crews of evacuation vehicles.

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Exposure of Emergency Workers

In an emergency exposure, the relevant **requirements for occupational exposure in planned exposures** are applied for emergency workers, in accordance with a graded approach. Response organizations and employers ensure that no emergency worker, other than those whose perform exceptional tasks, is subject to an exposure in an emergency in excess of **50 mSv**.

50

Guidance Values for Limiting Exposure of Emergency Workers

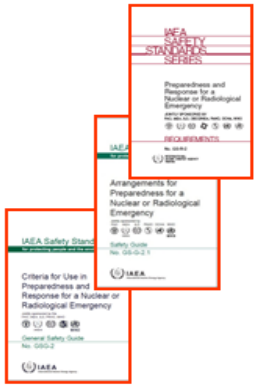
Action	$H_P(10)$
Life saving	< 500 mSv(*)
To prevent severe deterministic health effects To prevent development of catastrophic conditions	< 500 mSv
To avert a large collective dose	< 100 mSv

(*) This value may be exceeded under circumstances in which the benefit to others clearly outweighs the emergency worker's own risk and the emergency worker volunteers to take the action, understands and accepts this risk.

51

IAEA Safety Standards

1. **Preparedness and Response for a Nuclear or Radiological Emergency: Requirements GS-R-2 (2002)**
2. **Arrangements for Preparedness for a Nuclear or Radiological Emergency: Guidance GS-G-2.1 (2007)**
3. **Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency: Guidance GSG-2 (2011)**



52

IAEA Emergency Preparedness and Response Series

EPR-EMBARKING 2012

Considerations in Emergency Preparedness and Response for a State Embarking on a Nuclear Power Programme

53

Most recent EPR publication of IAEA

Actions to Protect the Public in an Emergency due to Severe Conditions a a Light Water Reactor

(EPR-NPP PUBLIC PROTECTIVE ACTIONS, 2013)

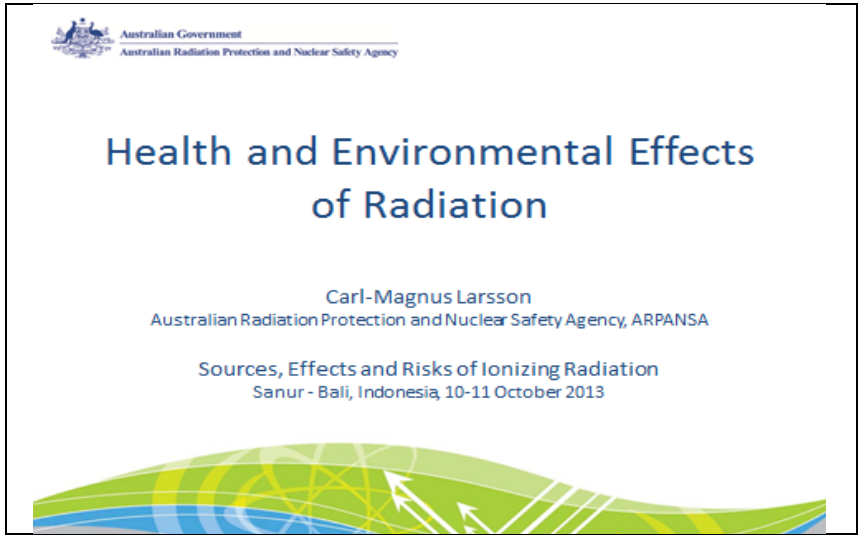
54

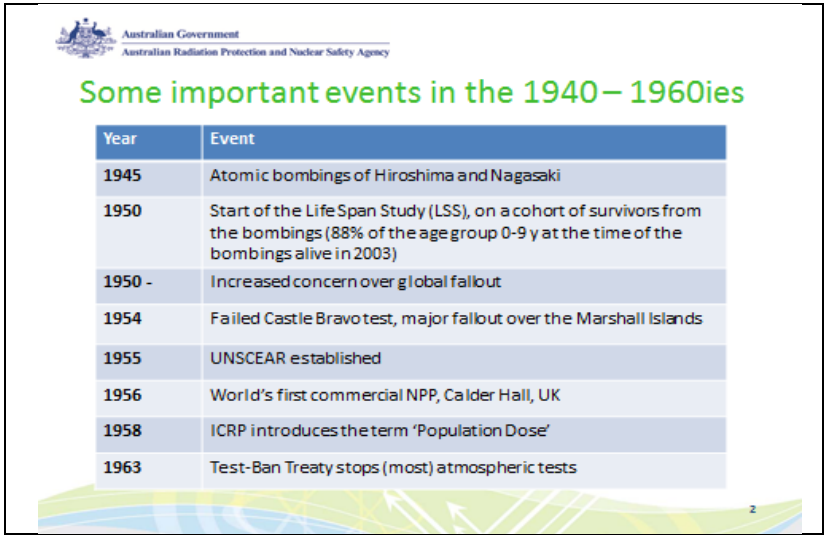
INVITED SPEAKER :

Dr. Carl Magnus Larsson,

ARPANSA

Australia





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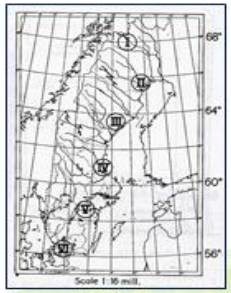
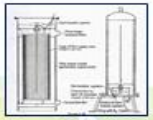
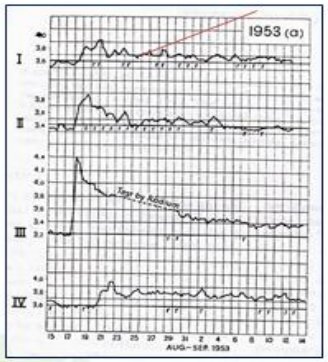
Measurement of fallout in Sweden 1953

Test at Novaja Zemlja 12 August 1953



Radiation levels increase from 17 August





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Rolf Sievert writes to Swedish PM 4 April 1954

"Large-scale tests of nuclear weapons in the western parts of the SSSR could under unfavourable weather conditions undoubtely cause serious radiation injuries in the Swedish population"

Rolf Maximilian Sievert
1896 - 1966

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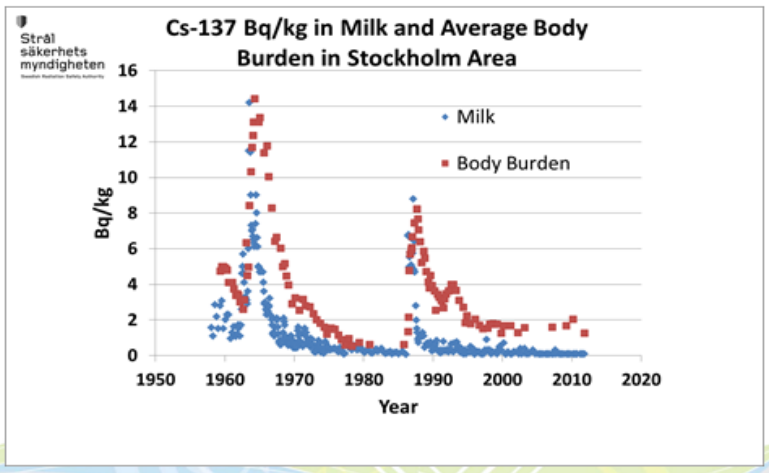

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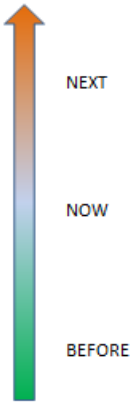
UNSCEAR - recent developments

Radiation exposure from electricity generation
Updated methodology for estimating human exposures due to discharges
Biological effects from selected internal emitters
Epidemiology of low-dose-rate to the public

- Levels and effects of radiation exposure due to the nuclear accident following the 2011 Great East-Japan Earthquake and Tsunami
- Effects of radiation exposure of children
- Attributing health effects to ionizing radiation exposure and inferring risks
- Uncertainties in risk estimates for cancer due to exposure to ionizing radiation

Summary of low-dose health effects and white paper on biological mechanisms at low doses
2008 Scientific Annexes
2006 Scientific Annexes



6

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Issues covered in my presentation

- Does recent studies of paediatric CT change our understanding of
  - cancer risk at *low doses*?
  - use of *paediatric CT*? and,
  - cancer risk for *children in general*?
- Environmental effects; does Fukushima shed any further light?

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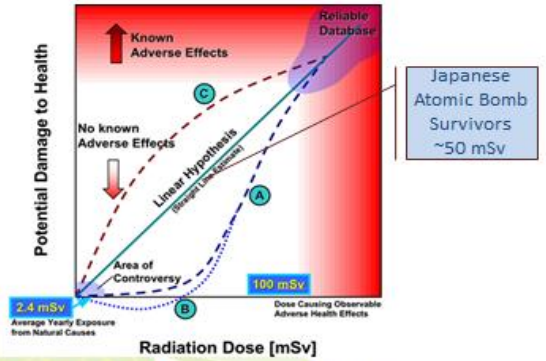
Linear No Threshold Hypothesis



http://www.yomiuri.co.jp/adv/wol/photo/opinion/earthquake_110418_04_cn.jpg

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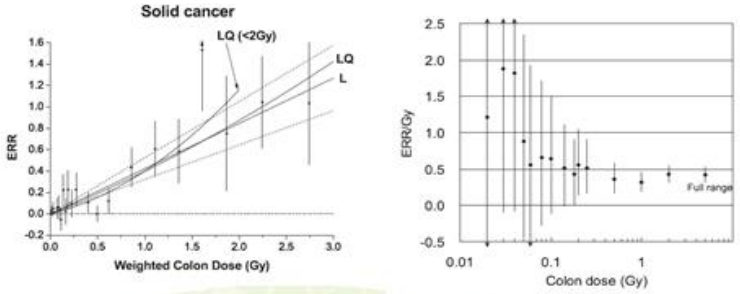
LSS Report 14, 1950 - 2003[1]



[1]Ozasa et al., Rad. Res. 177, 229–243 (2012)

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LSS Report 14, 1950 - 2003[1]

"The estimated lowest dose range with a significant ERR for all solid cancer was 0 to 0.20 Gy, and a formal dose-threshold analysis indicated no threshold; i.e., zero dose was the best estimate of the threshold."

[1]Ozasa et al., Rad. Res. 177, 229–243 (2012)

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Cancer risk from paediatric CT

THE UNIVERSITY OF MELBOURNE

MELBOURNE SCHOOL OF POPULATION HEALTH

John Mathews et al. BMJ 2013 (incl. Anthony Wallace, ARPANSA)

Cancer risk in 680 000 people exposed to computed tomography scans in childhood or adolescence: data linkage study of 11 million Australians

11

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A large study

CT exposed

680,211

- Exposure more than 12 months prior to any cancer diagnosis
- When aged 0-19 years
- In period 1985-2005
- Follow-up to 31/12/2007

Non-exposed

10,259,469

- No Medicare record of any CT scan
- When aged 0-19 years
- In period 1985-2005
- Follow-up to 31/12/2007

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Number of cancers of all types and IRRs by lag period

	Lag period		
	1 year	5 years	10 years
Exposed cohort			
No of observed cancers	3130	2365	1405
No of person years	6 486 548	3 971 641	1 808 883
Mean years of follow-up	9.5	7.3	5.5
Expected no of cancers*	2542	1963	1196
Excess no of cancers	608	402	209
Unexposed cohort			
No of observed cancers	57 524	58 309	59 269
No of person years	177 191 342	179 706 249	181 869 007
Mean years of follow-up	17.3	17.3	17.1
IRR (95% CI; exposed v unexposed)	1.24 (1.20 to 1.29)	1.21 (1.16 to 1.26)	1.18 (1.11 to 1.24)
χ^2 (1 df) for departure of IRR from unity	129.1	74.4	33.8
P for departure of IRR from unity	P<0.001	P<0.001	P<0.001

*Calculated from rates in unexposed people after stratification for age, sex, and year of birth.

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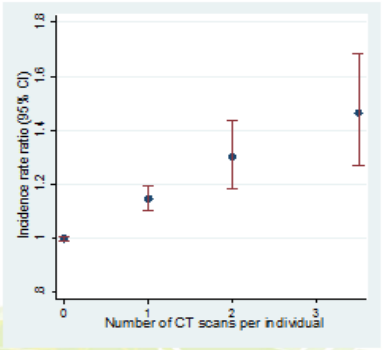
Incidence rate ratios (IRR), exposed versus unexposed, for all cancers and 95% confidence intervals (CI) based on a 1 year lag period, by number of CT scans



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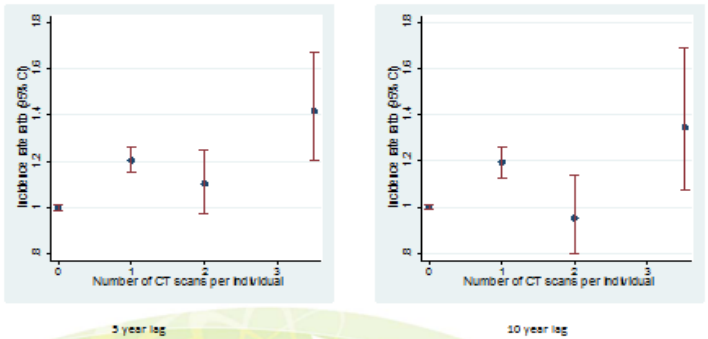
Incidence rate ratios (IRR), exposed versus unexposed, for cancers of all types and 95% confidence intervals(CI) by number of CT scans, based on 5 and 10 year lag periods.



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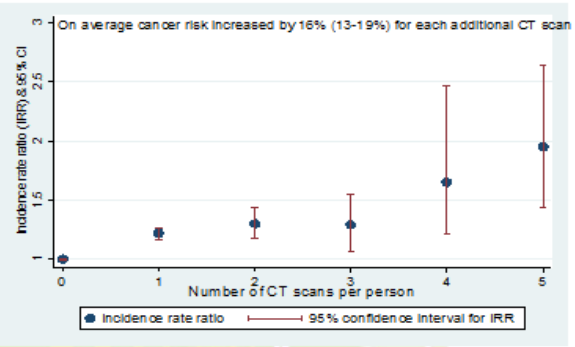
Risk increases with number of scans



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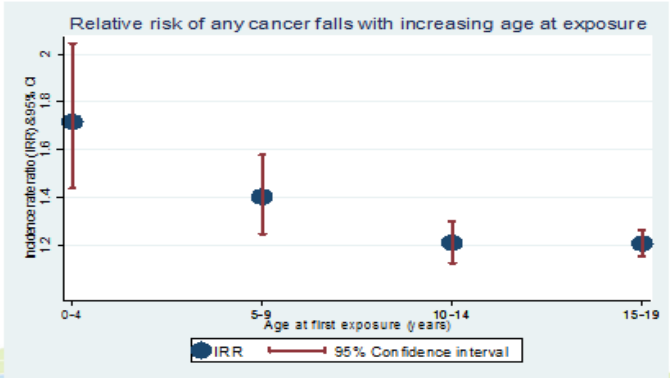
Risk by age of exposure – all cancer



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Australian Radiation Protection and Nuclear Safety Agency

Observations

- Results broadly consistent with LNT and risk factors as estimated in BEIR VII and by UNSCEAR
- Corresponds to leukaemia and brain cancer data from the UK (Pearce et al. 2012)
- Results indicate a significant increase in what still constitutes a very small risk
- A cancer attributable to CT between 1 in every 1400 and 1 in every 2000 scans (4.5 – 2 mSv).

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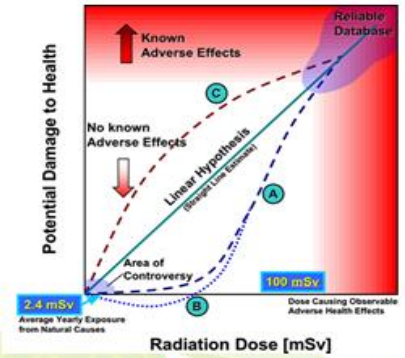
Linear No Threshold Hypothesis



http://www.yomiuri.co.jp/adv/wol/photo/opinion/earthquake_110415_04_cn.jpg

19

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Observations

- The increase in cancer in this cohort is max 1% of all cancers; however, it is confined to a sub-set of 6% of the cohort
- Thus, the findings reinforce the need for justification of CT procedures on children

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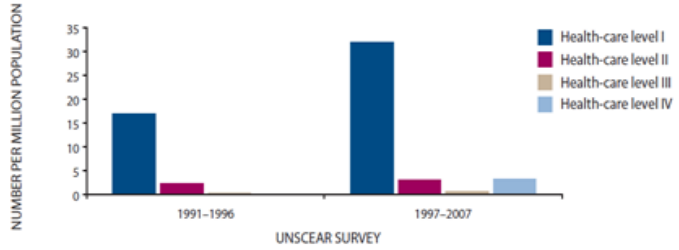
Provision of CT Scanners[1]



[1]UNSCEAR 2008 Annex A

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Age Dependent Risk

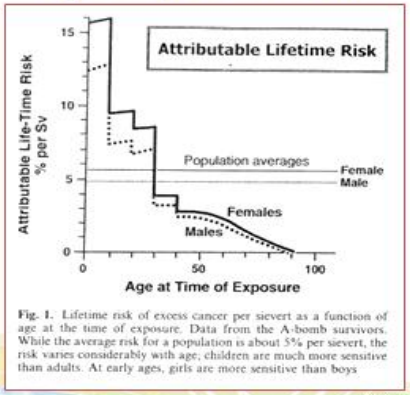

Fig. 1. Lifetime risk of excess cancer per sievert as a function of age at the time of exposure. Data from the A-bomb survivors. While the average risk for a population is about 5% per sievert, the risk varies considerably with age; children are much more sensitive than adults. At early ages, girls are more sensitive than boys

Hall, Pediatric Radiology, (2002) Vol.32

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UNSCEAR 'Children report'[1]

- 23 cancer types in children compared to adults following radiation exposure
  - Higher for 25% (e.g. leukaemia, breast, thyroid skin, brain
  - About the same in 15% (e.g. colon)
  - Less in about 10% (e.g. lung)
  - Weak or no association to radiation (in children and adults) in 30%
  - Data inconclusive in 20%

[1]UNSCEAR 2013 Annex B

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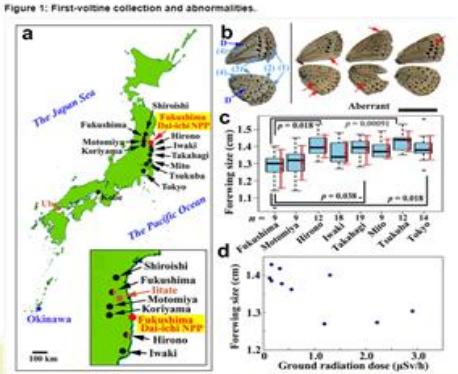
Environmental effects

Erica tetralix L.

Australian Government
Australian Radiation Protection and Nuclear Safety Agency

The pale grass blue butterfly[1]



Hiyama et al. 2012

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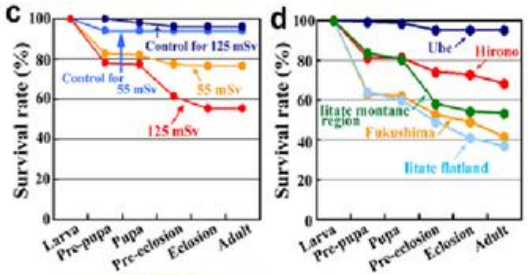
The pale grass blue butterfly[1]



Hiyama et al. 2012

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Species Sensitivity Distributions, SSDs

Treatment of all experimental data, following QA, to derive predicted no-effect-dose-rate values for ecosystems

Building on methodology developed for chemical hazards


ERICA
(Contract Number: FI6R-CT-2004-508847)

DELIVERABLE 5:

Derivation of Predicted-No-Effect-Dose-Rate values for ecosystems (and their sub-organisational levels) exposed to radioactive substances

Erica tetralix L.

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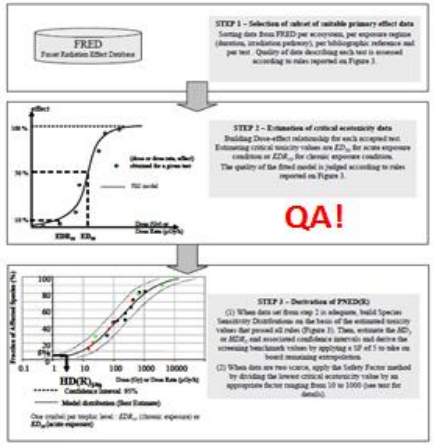
SSD Procedure



Erica tetralix L.

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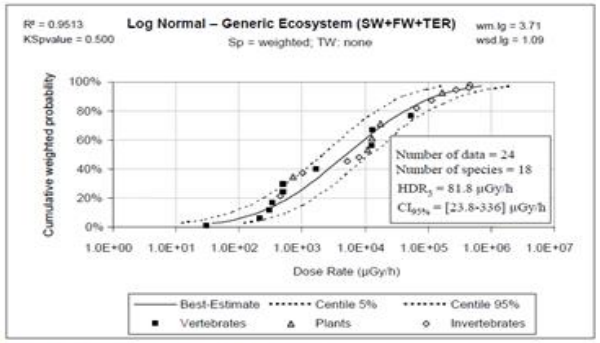
Species Sensitivity Distribution

Erica tetralix L.



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Outside the 5% + 95% CI range

Erica tetralix L.

Chronic external †					EDR_{10} (eGy/h)
Generic (FW)	Fish (V)	*Oryzias latipes* (medaka)	Reproduction	Ovary weight at 70 days of age	24.9
				Testis weight at 70 days of age	6.7
Generic (TER)	Mammals (V)	*Mus musculus* (mouse)	Reproduction	Germ cells per ovary at 56 days of age	195.6
				N° of litters per fertile female during 245 days (mean; SE).	26.2
		Rattus norvegicus (rat)	Reproduction	A1 spermatogonia (% of control)	23.8
		Capra hircus (goat)	Reproduction	Total sperm production (% of control)	11.6
Generic (SW)	Annelids (I)	*Neanthes arenaceodentata*	Reproduction	% Abnormal embryos % of broods with <25% abnormal embryos	1.44
			Reproduction	% live embryos % of broods with >75% embryos	134
	Fish (V)	*Pleuronectes platessa*	Reproduction	Mean proportion of plaice testes occupied by different cell types irradiated for 197 days - sperm	53.4
			Reproduction	Mean proportion of plaice testes occupied by different cell types irradiated for 73 days - non germal cells	193
			Reproduction	Mean proportion of plaice testes occupied by different cell types irradiated for 73 days - spermatogonia	193

† Trophic Level: I invertebrates; V vertebrates and P plants

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Environmental Effects

- Robust experimental evidence for PNEDR in the µGy h^{-1} range – but species dependent
- Such values relevant for protection purposes
- Population and ecosystem effects at such levels maybe less likely
- But data gaps as regards species, and incomplete understanding of effects at higher levels of biological organisation

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THANK YOU

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INVITED SPEAKER :
Dr. Soehartati Gondhowiardjo,
Cipto Mangunkusumo Hospital (RSCM),

Indonesia

CURRICULUM VITAE

Soehartati Gondhowiardjo, MD. PhD

Born	: Jakarta, September 1955
Education	
1. GP	: FKUI, 1980
2. Radiologist	: FKUI, 1987
3. Radiation Oncologist	: FKUI / Muenster (Germany) 1990
4. PhD	: FKUI, 1998
	(EBV LMP1 and Proliferation in NPC)

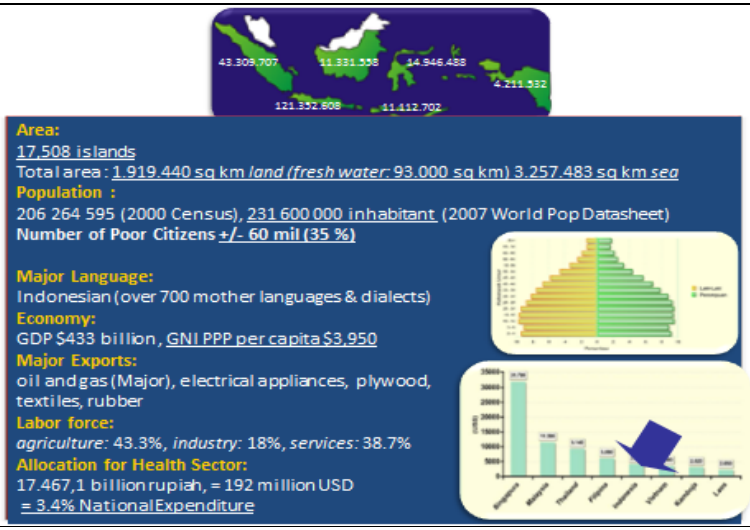
Current Positions :
Head of Radiotherapy Department , RSUPN-CM / FKUI
Chairman of Indonesian Radiation Oncology Society
Past President of South East Asia Radiation Oncology Group

Current Status Medical Radiation Exposures in Indonesia :

Radiation Safety in Radiotherapy as a model

Soehartati Gondhowiardo, MD, PhD
Henry Kodrat MD
Gregorius BP MD

Department of Radiotherapy
Cipto Mangunkusumo National General Hospital, Faculty of Medicine, University of Indonesia



Overview

- Medical Radiation Exposure
  - Radiodiagnostic
  - Nuclear Medicine
  - Radiotherapy data
  - Others (Laboratory and Research)
  - Extrapolation data of Radiology / medical exposure in Indonesia
- Radiotherapy Status in Indonesia
  - Back ground
  - Protection From Risk
  - Milestones in Radiotherapy
- Take Home messages

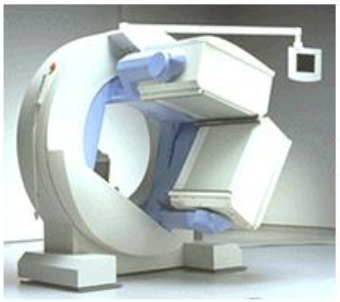
Diagnostic Imaging

- Conventional X-ray
- Flouroscopy
- CT-Scan

- **Every hospital has a radiology service**
- **Almost type B hospital has a CT-Scan**
- **No of Hospital ….(2012)**
  - **A … 46 + 6**
  - **B … 196 + 34**
  - **C … 418 + 96**
  - **D … 238 + 75**

–TOTAL … 1864

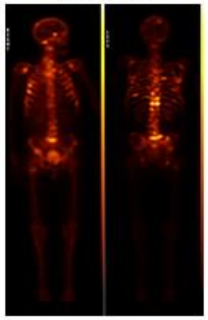
Nuclear Medicine



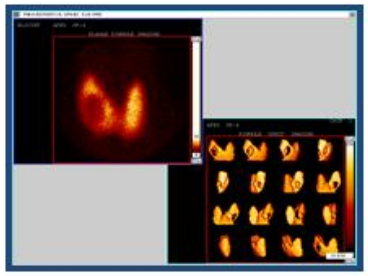
- Nuclear imaging detects **functional (vs. anatomical)** properties of the human tissue.
- The imaging is done by tracing the **distribution of radiopharmaceuticals** within the body with a gamma camera

Nuclear Medicine

Bone Scan

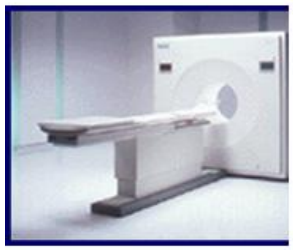


Thyroid Scan



PET/CT-Scan

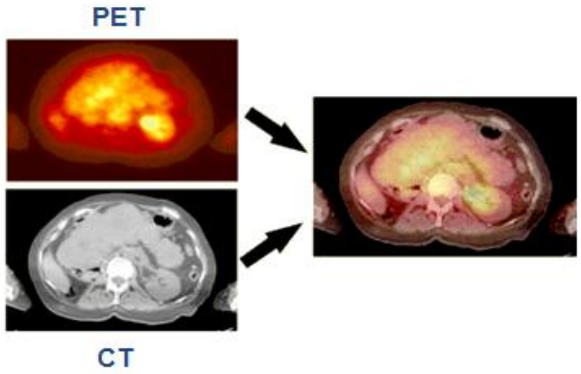
PET Scanner



Cyclotrone reactors

- 3 PET scanners in Indonesia in Jakarta
- 3 Cyclotrons reactors

PET/CT-Scan



Nuclear Medicine as a Therapeutic Modality

Radiopharmaceutical		For treatment of	Route of administration	Maximum activity
I-131	iodide	Thyrotoxicosis	Oral	1 GBq
I-131	iodide	Carcinoma of thyroid	Oral	20 GBq
I-131	MIBG	Malignancy	IV	10 GBq
P-32	phosphate	Polycythaemia vera	IV or oral	200 MBq
Sr-89	chloride	Bone metastases	IV	150 MBq
Y-90	colloid	Arthritic conditions/ malignant effusions	Intra-articular Intra-cavitary	250 MBq 5 GBq
Y-90	spheres	Hepatocellular Carcinoma	Intra-articular	100 MBq
Er-169	colloid	Arthritic conditions	Intra-articular	50 MBq
Re-186	colloid	Arthritic conditions	Intra-articular	150 MBq

- **15 center** with Nuclear Medicine Medical Services .....
- Jakarta (7),
- Bandung (1)
- Medan (1), Padang (1),
- Jogyakarta (1), Semarang (1),
- Malang (1), Surabaya (1),
- Makasar (1)

Oopss ... we have no other accrual data

Terminology

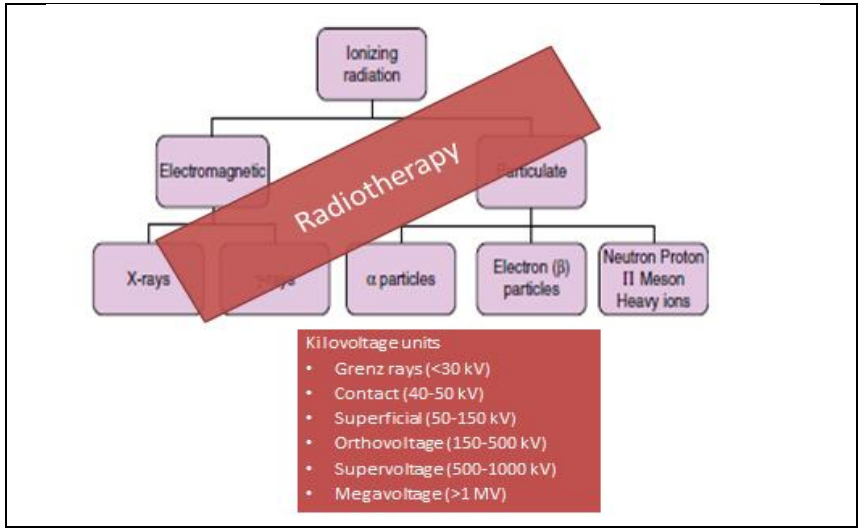
Radiation oncology is that discipline of human medicine concerned with the generation, conservation, and dissemination of knowledge concerning the causes, prevention, and treatment of cancer and other diseases involving special expertise in the therapeutic applications of ionizing radiation.

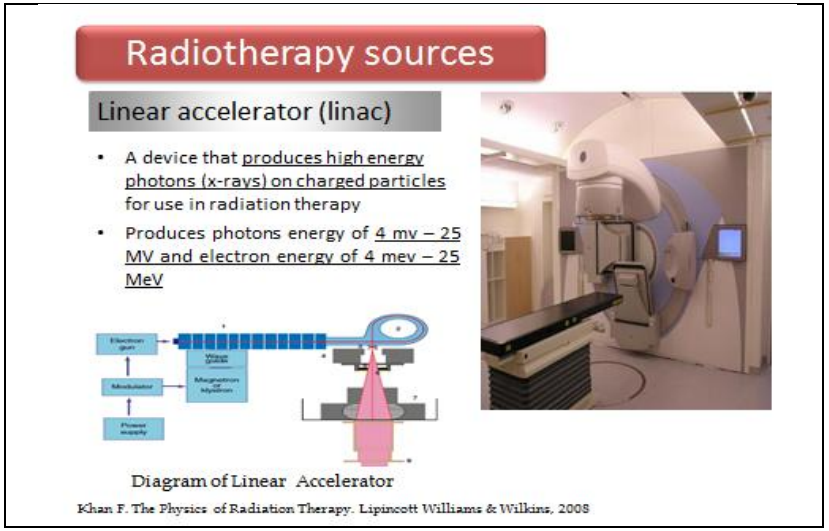
Radiation therapy is a clinical modality dealing with the use of ionizing radiations in the treatment of patients with malignant tumor (and occasionally benign diseases).

Radiation oncologists use radiation therapy to try to cure cancer, to control cancer growth or to relieve symptoms, such as pain.

Knowledge	• Radiation Oncology
Modality	• Radiotherapy
Person	• Radiation Oncologist

Halperin EC et al. Principles and Practices of Radiation Oncology Lipincott Williams & Wilkins, 2008


Ionizing radiation
Electromagnetic
Particulate
Radiotherapy
X-rays
α particles
Electron (β) particles
Neutron Proton
II Meson
Heavy ions
Kilovoltage units
• Grenz rays (<30 kV)
• Contact (40-50 kV)
• Superficial (50-150 kV)
• Orthovoltage (150-500 kV)
• Supervoltage (500-1000 kV)
• Megavoltage (>1 MV)


Radiotherapy sources
Linear accelerator (linac)
• A device that produces high energy photons (x-rays) on charged particles for use in radiation therapy
• Produces photons energy of 4 mv – 25 MV and electron energy of 4 mev – 25 MeV
Diagram of Linear Accelerator
Khan F. The Physics of Radiation Therapy. Lipincott Williams & Wilkins, 2008

Telecobalt-60

- A devices that use cobalt-60 as the source of radiation
- Cobalt-60 unit have a cylindrical source 2 cm in diameter
- Produces gamma rays energy of 1.17 – 1.35 mv
- The half-life of Co-60 is 5,27 years

Source !!!

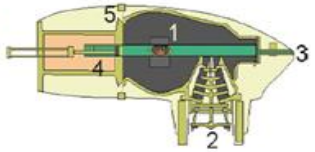


Diagram of Cobalt-60 treatment head

Beyzadeoglu M et al. Basic Radiation Oncology. Springer, 2008

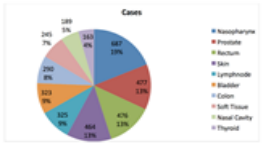
the fact .....??

- "The proportion of patients with cancer in whom external beam radiotherapy is indicated based on the best available evidence was calculated to be ***52%***"
- "Radiotherapy is widely known to be one of ***the safest areas of modern medicine***" ... ?
- ....."yes, for some, ***this essential treatment can bring harm, personal tragedy and even death***"

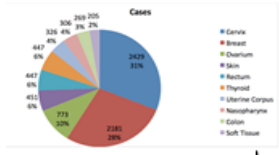
Delaney, Cancer 2005
Radiotherapy Risk Profile, World Health Organization 2008

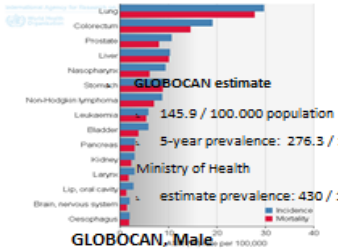
BASIC CANCER INFORMATION

Most Common Male Cancers – Pathology Based Cancer Registry

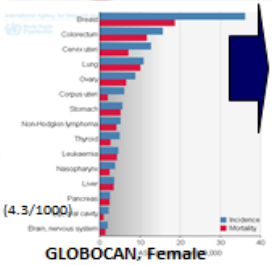


Most Common Female Cancers – Pathology Based Cancer Registry





GLOBOCAN, Male



GLOBOCAN, Female

GLOBOCAN estimate
145.9 / 100.000 population
5-year prevalence: 276.3 / 100.000 population

Ministry of Health
estimate prevalence: 430 / 100.000 population (4.3/1000)

Optimal Radiotherapy Utilization Rate by Cancer Type

Tumor type	Proportion of all cancers	Proportion of patients receiving radiotherapy	Patients receiving radiotherapy (% of all cancers)
Breast	0.13	83	10.8
Lung	0.10	76	7.6
Melanoma	0.11	23	2.5
Prostate	0.12	60	7.2
Gynecologic	0.05	35	1.8
Colon	0.09	14	1.3
Rectum	0.05	61	3.1
Head and neck	0.04	78	3.1
Gall bladder	0.01	13	0.1
Liver	0.01	0	0.0
Esophageal	0.01	80	0.8
Stomach	0.02	68	1.4
Pancreas	0.02	57	1.1
Lymphoma	0.04	65	2.6
Leukemia	0.03	4	0.1
Myeloma	0.01	38	0.4
Central nervous system	0.02	92	1.8
Renal	0.03	27	0.8
Bladder	0.03	58	1.7
Testis	0.01	49	0.5
Thyroid	0.01	10	0.1
Unknown primary	0.04	61	2.4
Other	0.02	50	1.0
Total	1.00	-	52.3

Delaney, 2005

Indications for Radiotherapy in Patients with Breast Carcinoma: Levels and Sources of Evidence

Outcome no.	Clinical scenario	Treatment Indicated	Level of evidence*	Reference(s)	Proportion of all breast carcinoma cases
1	DCIS, breast-conserving surgery	Adjuvant radiotherapy	II	NHMRC National Breast Cancer Centre,[7] NCI,[13] NCCN,[14] Cancer Care Ontario Practice Guideline Initiative,[10] BC Cancer Agency,[15] The Steering Committee on Clinical Practice Guidelines for the Care and Treatment of Breast Cancer[16]	0.09
2	DCIS treated with mastectomy, local recurrence	Radical or adjuvant radiotherapy	III	NHMRC National Breast Cancer Centre,[7] NCI,[13] NCCN,[14]	
4	T1–T2, N0–N1, M0, breast-conserving surgery	Adjuvant radiotherapy	II		
5	T1–T2, N0–N1, M0 mastectomy			DeVita et al (1979) as quoted in Van Der Kogel et al (2009)	0.01
6	"Local treatment, which includes surgery or radiotherapy, could be expected to be successful in ... 40% of these cases. In perhaps 15% of the total, radiotherapy would be the principal form of treatment."		I	NHMRC Radiation Oncology Advisory Group,[10] NCI,[13] NCCN,[14] NIH Consensus Development Panel[17]	0.03
13		Radiotherapy +/– systemic therapy, +/– surgery	III	NHMRC National Breast Cancer Centre,[7] Recht et al,[12] NCCN,[14] BC Cancer Agency[15]	0.06
6, 17	...any T, any N, M1, painful bone metastases or T1–T2, N0–N1, M0, distant recurrence with painful bone metastases	Palliative radiotherapy	I	NHMRC National Breast Cancer Centre,[7] NCI,[13] ASCO,[11] Cancer Care Ontario Practice Guideline Initiative,[10] BC Cancer Agency,[15] Scottish Intercollegiate Guidelines Network[18]	0.01
9, 18	Any T, any N, M1, brain metastases or T1–T2, N0–N1, M0, distant recurrence with symptomatic brain metastases	Palliative radiotherapy +/– surgery	II	NHMRC National Breast Cancer Centre,[7] NCI,[13] Simpson et al,[11] Loeffler et al,[19] BC Cancer Agency,[15] Scottish Intercollegiate Guidelines Network[18]	< 0.01
Proportion of all patients with breast carcinoma for whom radiotherapy is indicated					0.03

52.3% cancer patients have indication for RT

RI ... 1 / 1000 new cases or 4.3/ 1000 prevalence

Delaney, 2003

Radiotherapy..... Distribution of Cases

No	Site	Proportion (%)
1	Cervix	25.2%
2	Breast	20.7%
3	Nasopharynx	14.8%
4	Central Nervous System	9.7%
5	Lung and Bronchial Tree	5.2%
6	Lymphnodes	3.0%
7	Bone and Joints	2.7%
8	Colorectal	2.6%
9	Thyroid	2.4%
10	Larynx	1.7%
11	Prostate	
12	Skin	
13	Others	
TOTAL		

7/

> 52.3% .. ?

IMRT in RSCM → 15 % (+ breast etc)

Why radiotherapy is important?

An ***estimated 5.1 million courses*** of radiotherapy treatment were administered annually between 1997 and 2007 vs ***4.3 million*** in 1988

50-60% of cancer patients could benefit from radiation therapyThe fraction of cancer patients treated with RT is increasing,

* UNSCEAR 2008 Report

Megavoltage Units per Million Population, 2012

41 equipment for 250 million (24 linac) population

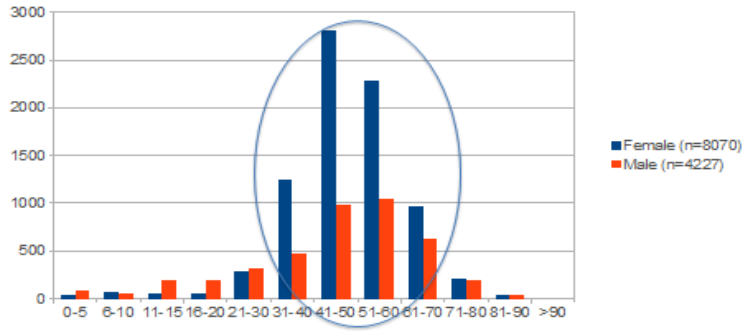
EXISTING RADIOTHERAPY PATIENTS REFERAL SERVICES

13 RT equipment for 9.7 million population
But ..... 63.5 % of our patient come from outside of Jakarta

Indications for radiotherapy

No	Radiotherapy Indication	Proj. Annual Treatments
1	Palliative Treatments	**3570**
2	Gynaecological tumour	3065
3	Head/neck	2214
4	Breast	1561
5	Brain	561
6	Bone and other sarcomas	239
7	Lymphomas	231
8	Colorectal	181
9	Lung	148
10	Skin	**107**

Gender and Age



In children, more male receive RT vs female
This is reversed in adults where more females receive RT

Doses and Fractionation

No	Radiotherapy Indication	Dose/ Gy	Typical Fract
1	Palliative Treatments	**8-40 (30)**	**1-20 (10)**
2	Gynaecological tumour	50	25
3	Head/neck	**60-70(66)**	**30-35(33)**
4	Breast	50-60(50)	25-30(25)
5	Brain	54-60(54)	27-30(27)
6	Bone and other sarcomas	50-66(60)	25-33(30)
7	Lymphomas	20-50(40)	10-25(20)
8	Colorectal	25-54(25)	5-27(5)
9	Lung	50-66(60)	25-33(30)
10	Skin	60-66(60)	30-33(30)

Doses and Fractionation

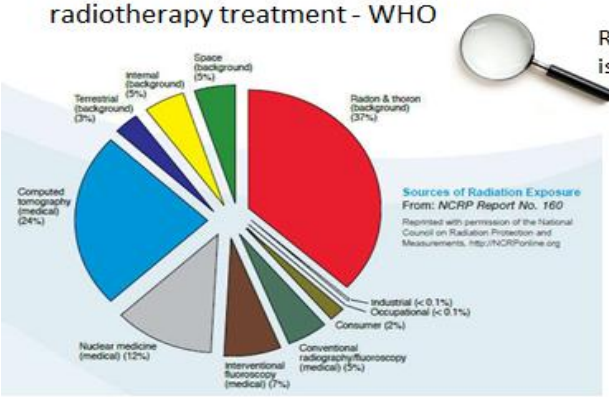
No	Radiotherapy Indication	Proj. Annual Dose (Gy)	Proj. Annual Fract
1	Palliative Treatments	107100	35700
2	**Gynaecological tumour**	**153250**	**76625**
3	Head/neck	146124	73062
4	Breast	78050	39025
5	Brain	30294	15147
6	Bone and other sarcomas	14340	7170
7	Lymphomas	9240	4620
8	**Colorectal**	**4525**	**905**
9	Lung	8880	4440
10	Skin	6420	3210

Medical Exposure from Radiotherapy

- **Large doses of radiation** is used annually but only to a **small number of patients** due to limited coverage. (nearly half a million Gray to a total of **12,286 patients** reported to receive RT in Indonesia in 2012)
- **Different from radiology and nuclear medicine where significantly larger number of patients receive significantly smaller doses each.**
- Different implications for recording and reporting exposures
- Note: this has not taken into account brachytherapy

Annually worldwide,
more than 3600 million x-ray exam performed, 3,7 million nuclear medicine done, and 7,5 million radiotherapy treatment - WHO

Radiotherapy is too small



Annually worldwide,
more than 3600 million x-ray exam performed, 3,7 million nuclear medicine done, and 7,5 million radiotherapy treatment - WHO

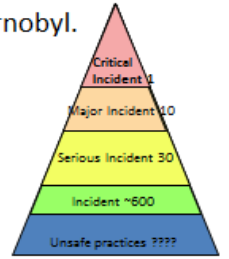
- Radiotherapy in Indonesia 12.000 treatments
- How many procedure in diagnostic imaging and nuclear medicine?
- Nuclear Medicine = **5,919 procedures???**
- Diagnostic Imaging = **5,760,000 procedures???**
- Opps how is the **INDICATION** ...?

Overview

- Medical Radiation Exposure
  - Radiodiagnostic
  - Nuclear Medicine
  - Radiotherapy data
  - Others (Laboratory and Research)
  - Extrapolation data of Radiology / medical exposure in Indonesia
- **Radiotherapy Status in Indonesia**
  - **Back ground**
  - **Protection From Risk**
  - **Milestones in Radiotherapy**
- Take Home messages

Background.....

- Over the last three decades, at least 3000 patients have been affected by radiotherapy **incidents and accidents**.
- **more** acute radiation deaths than any other causes , including Chernobyl.
- do not only affect ***patients*** directly (e.g. harm and death), but might also undermine ***the public's*** confidence in the treatment.



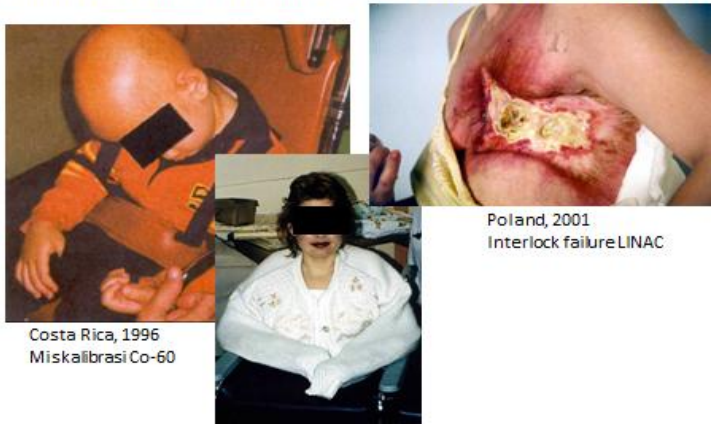
How does the general public perceive radiotherapy?

- "…when radiotherapy does hit the headlines the story is about the ***very rare occasions*** when there have been mistakes."
- "…The machinery was so **daunting** I had imagined everything from **James Bond to Aliens** …"
- "…Despite its effectiveness, the image of radiotherapy has been **stigmatised** … wrongly assume they will be getting ***second best*** and will suffer lots ***of unpleasant side-effects.***"

Catherine Woolams http://www.canceractive.com/cancer-active-page-link.aspx?n=260
BBC News http://www.bbc.co.uk/news/health-11161653

The problem:

.....accidents might be rare, but terrible when happen



Costa Rica, 1996
Miskalibrasi Co-60

Poland, 2001
Interlock failure LINAC

Publication …. ????

Safety in radiotherapy is crucial

Not only in the past, and now the press is more active

THE RADIATION BOOM

Radiation Offers New Cures, and Ways to Do Harm

By WALT BOGDANICH
Published: January 23, 2010

As Scott Jerome-Parks lay dying, he clung to this wish: that his fatal radiation overdose — which left him deaf, struggling to see, unable to swallow, burned, with his teeth falling out, with ulcers in his mouth and throat, nauseated, in severe pain and finally unable to breathe — be studied and talked about publicly so that others might not have to live his nightmare.

RECOMMEND
TWITTER
LINKEDIN
SIGN IN TO E-MAIL
PRINT
REPRINTS
SHARE

Enlarge This Image

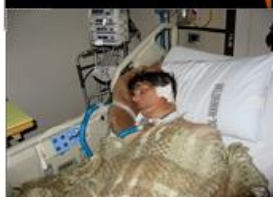
For his last Christmas, Scott Jerome-Parks rested his feet in buckets of sand his friends had sent from a childhood beach. More Photos »

Sensing death was near, Mr. Jerome-Parks summoned his family for a final Christmas. His friends sent two buckets of sand from the beach where they had played as children so he could touch it, feel it and remember better days.

Mr. Jerome-Parks died several weeks later in 2007. He was 43.

A New York City hospital treating him for tongue cancer had failed to detect a computer error that directed a linear accelerator to blast his brain stem and neck with errant beams of radiation. Not once, but on three consecutive days.

New york, 2005
Data transfer error resulting in failure to transmit MLC control point



Let's look at the numbers:……..
How safe is radiotherapy?

Outcome	Activity, intervention, or group	*n/million*	Units
Fatal cancer	Whole-body spiral CT	480	Examinations
Death	**medical error**	**1310**	**Admissions to U.S. hospitals**
Death	U.K. motorcyclists	1138	Motorcycles/y
Killed	U.K. car users	78	Cars/y
Fatal crash	General aviation (noncommercial flights)	13	Flight hours
Fatal crash	Commercial aviation	0,16	Flight hours
Crash	Scheduled flights	4	Departures

Williams, Int J Radiat Oncol Biol Phys 2011

Let's look at the numbers:……..
How safe is radiotherapy?

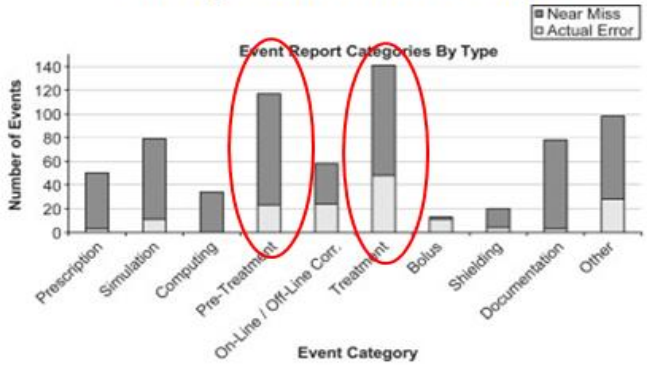
Outcome	Activity, intervention, or group	*n/million*	Units
Any error discovered after start of treatment	Canadian radiotherapy	19207	Courses of treatment
Error with significant dosimetric consequences (<20% variation)	U.S. radiotherapy	5500	Courses of treatment
Mild to moderate injurious outcome from radiotherapy errors	International review of safety in radiotherapy practice	1500	Courses of treatment
Error with "moderate" clinical consequences	Canadian radiotherapy	1031	Courses of treatment
Error with "severe" clinical consequences	Canadian radiotherapy	78	Courses of treatment
Death	**1% reported rate of death from adverse events in radiotherapy**	**15**	**Courses of treatment**

Williams, Int J Radiat Oncol Biol Phys 2011

The numbers summarized.....

- It **is more dangerous** to go to the hospital compared traveling by airplanes
- The risk of death from radiotherapy accidents is remarkably **lower** than from other medical errors
- When "4 per departure" is corrected with average passenger per flight (50-100) and fatal crash 2.5% of all crashes, the risk for death from radiotherapy accidents **(15/million course)** approximates death from airplane crashes **(10/million passenger)**
- However, even 1 death from accidental exposure should be prevented if possible. .....**How?**

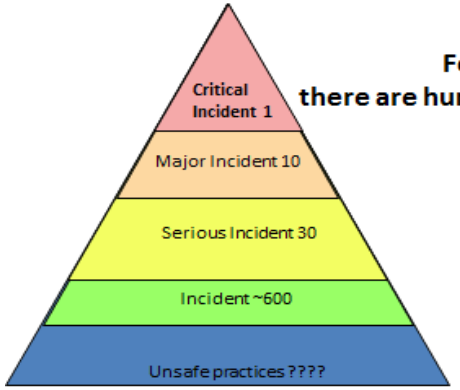
First, learn from events



Arnold A, Int J Radiat Oncol Biol Phys 2010

Underlying principle:

For every major accidents, there are hundreds "harmless" errors



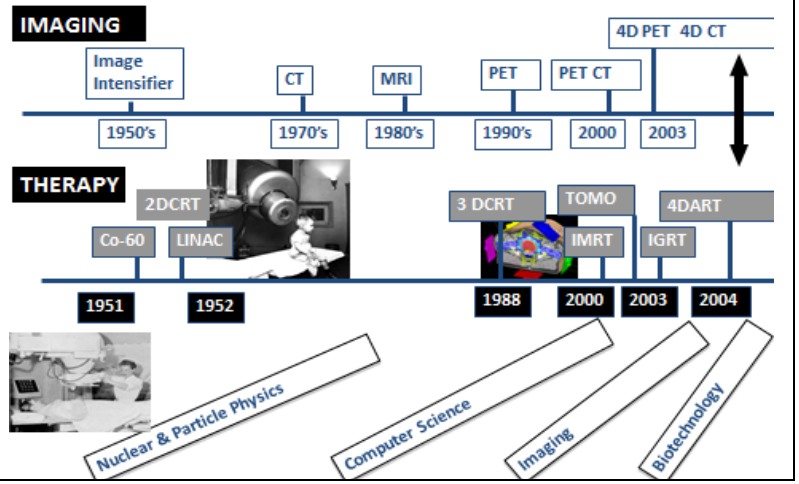
Errors will be perpetuated in later process ..... if no intervention is made

Protection from Risks

Lesson from accidental exposure with conventional techniques are available

They have been useful in improving safety and valid for new technologies, but accidents still happens

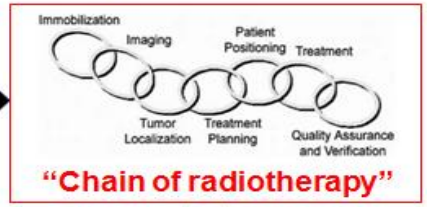
MILESTONES IN RADIOTHERAPY



Radiotherapy indication : .....Yes

Radiotherapy only or chemoradiation???

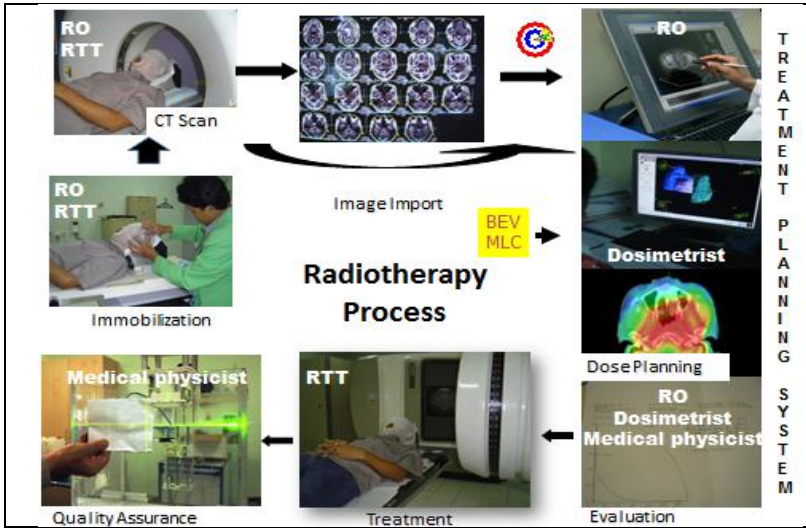
Out-patient Consultation

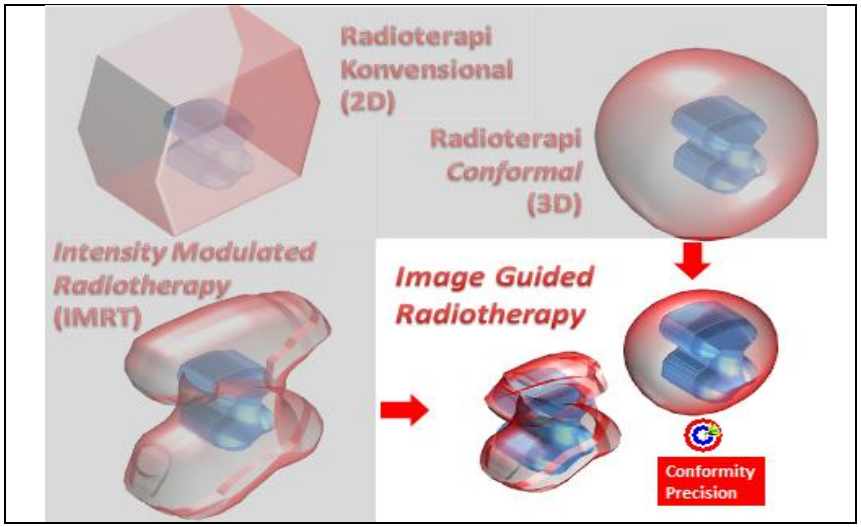


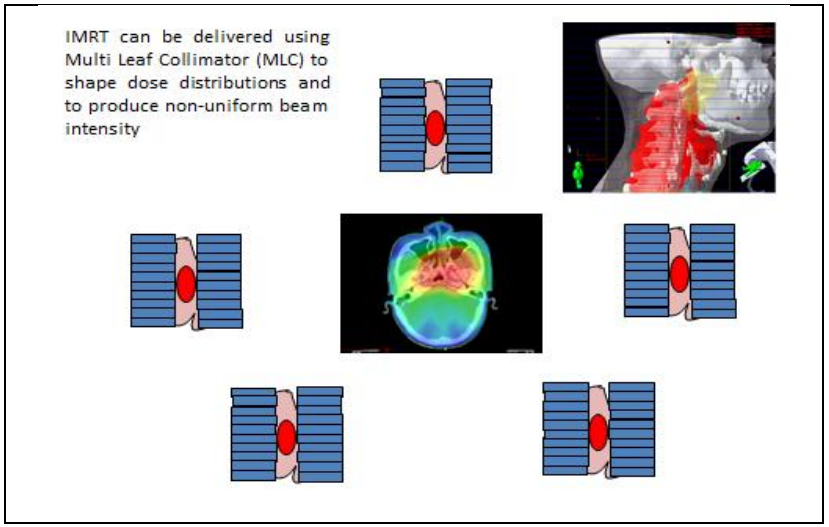
Follow-up ← No residual

Other modalities ← Residu (+)

Weekly consultation during radiation treatment




Radioterapi
Konvensional
(2D)
Radioterapi
Conformal
(3D)
Intensity Modulated
Radiotherapy
(IMRT)
Image Guided
Radiotherapy
Conformity
Precision


IMRT can be delivered using Multi Leaf Collimator (MLC) to shape dose distributions and to produce non-uniform beam intensity


Imaged Guided Radiotherapy

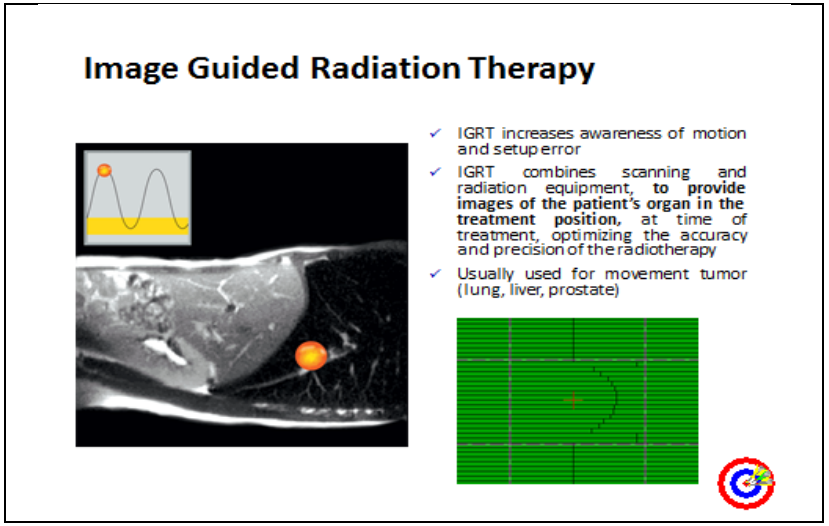
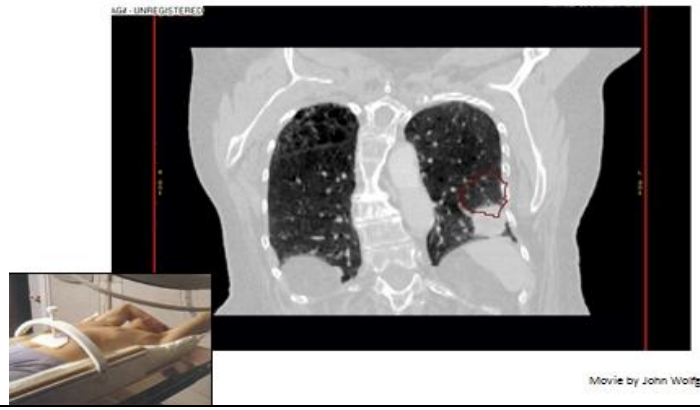


Image Guided Radiation Therapy
IGRT increases awareness of motion and setup error
IGRT combines scanning and radiation equipment, to provide images of the patient's organ in the treatment position, at time of treatment, optimizing the accuracy and precision of the radiotherapy
Usually used for movement tumor (lung, liver, prostate)

Image Guided Radiation Therapy

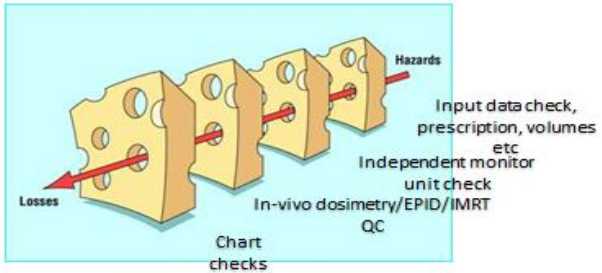


Overview

- Background
- Protection from risks
- Radiotherapy and its milestones
- **Take home messages**
  - **….. Need more exploration of medical exposure in the field of RD and NM**
  - **…… TR**

Radiotherapy Safety Layers

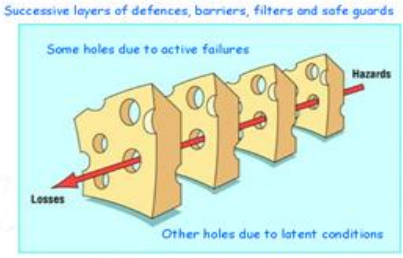
Successive layers of defences, barriers, filters and safe guards



When holes line up an error will occur

How to protect from errors?
Layers of prevention

Reason's Swiss Cheese Model of Failure Propagation

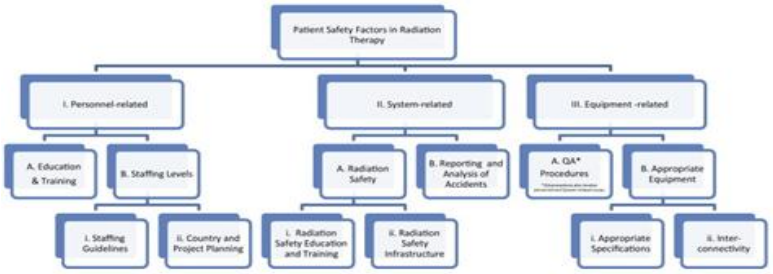


Safety culture

Continuous learning process

Courtesy from the IAEA

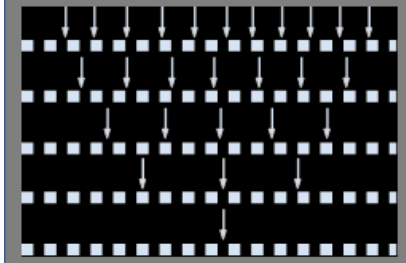
What are the components?



Abdel-Wahab M, Journal of the American College of Radiology 2011

MULTILAYERED ACCIDENT PREVENTION IN RT RSCM HOSPITAL

Initiating event – NEAR MISS (KNC)



Consequence – ACCIDENT (KTD)

MORNING MEETING 1
- TRAINING –EDUCATION
- COMMUNICATION

FRIDAY MEETING
- INDEPENDENT EXPERTICE
- COMMUNICATION

QA REVIEW
- INDEPENDENT REVIEW

•MORNING MEETING 2
- TRAINING _ EDUCATION
- REVIEW

CALIBRATION D/W/M/Y ETC

PREVENTIVE MAINTENANCE BY SCGEDULED

→ MEDICAL RECORD AUDIT

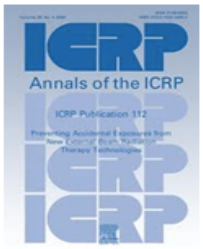
In Summary…

All must join hand in hand…
To provide best and safe treatment for the patients…

Courtesy of dr. Angela Giselvania

ICRP 112

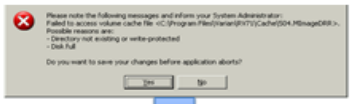
Preventing accidental exposure from new external beam radiation therapy technologies



- Purchasing new equipment without a concomitant effort on education and training and on a programme of quality assurance is **dangerous.**
- A programme for purchasing, acceptance testing and commissioning should not only address the treatment machine but also, **increasingly, complex treatment planning systems, "record and verify" systems, imaging equipment used for radiation therapy, software, procedures and entire processes.**
- There is a need for **re-commissioning** the relevant devices after equipment modifications or software updates and also to monitor impact on the related processes.

Incorrect IMRT planning/delivery !!!

- March 2005
- The patient begins an IMRT treatment at St. Vincent hospital, Manhattan, NY
- The plan had passed the QC process according to the local protocol
- Physicians want to modified dose distribution - replanning

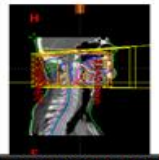


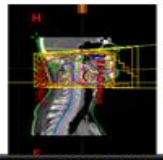
The transaction error message displayed

The New York Times

What happened? .....

- Monday - March 14, 2005, 1 p.m.
  - The patient is treated. The console screen would have indicated that MLC is not being used during treatment





The sagittal view should have looked like the one to the right, with MLCs

Introducing new technologies

- Decision to implement a new technology should be based on an evaluation of the expected benefit, rather than being driven by technology itself
- A step-by-step approach should be followed to ensure a safe implementation.

Staff training, availability and dedication

- Replacement of proper training with a short briefing or demonstration should be avoided, because important safety implications of new techniques cannot be fully appreciated from a short briefing.
- Certain safety-critical tasks, such as calibration, beam characterization, complex treatment planning and pretreatment verification, require a substantial increase in staff allocation.

Safety awareness of responsible persons for radiotherapy

- Independent verification of beam calibration remains essential.
- Investigating discrepancies in dose measurements before applying the beam to patient treatments.
- Hospital administrators of radiation therapy departments should provide a work environment that encourages working with awareness, facilitates concentration and avoids distraction.

Manufacturers

- Manufacturers should be aware of their responsibility for delivering the correct equipment with the correct calibration files and accompanying documents.
- They also have a responsibility to provide correct information and advice, upon request, from users.
- Procedures to meet these responsibilities should be developed and quality controlled.

Programme of purchasing, acceptance and commissioning

- Programmes for purchasing, acceptance testing and commissioning should not only address treatment machines but also treatment planning systems, radiation therapy information systems, imaging equipment used for radiation therapy, software, procedures and entire clinical processes.
- Devices and processes should be re-commissioned after equipment modifications including software upgrades and updates.

Need for new protocols for treatment prescription and dosimetry

- Protocols for treatment prescription, reporting and recording, such as found in ICRU reports, should be revised to accommodate new technologies.
- They should be adopted at a national level with the support of professional bodies. Similarly, dosimetry protocols should be developed for small and non-standard radiation fields.

Dose escalation

- Dose escalation without a concomitant increase in normal tissue complication probability generally implies a reduction of geometrical margins.
- Such a reduction is only possible with conformal therapy accompanied by precise, image-guided patient positioning and effective immobilization together with a clear understanding of the accuracy achieved in clinical practice.
- Without these features, target dose escalation could lead to severe patient complications

Safety-critical communication and notifications

- Unambiguous, well structured communication is essential, considering the complexity of radiation therapy and the multidisciplinary nature of the health care environment.
- In particular, procedures to notify physicists of maintenance and repair activities, identified as crucial in conventional technology, are even more necessary with new technologies.

Computers and data integrity

- Procedures should be in place to deal with situations created by computer "crashes", which may cause loss of data integrity and lead to severe accidental exposures.

Updating of quality control tests

- When conventional tests and checks are not applicable or not effective for new technologies, the safety philosophy should aim at finding measures to maintain the required level of safety.
- This may require the design of new tests or the modification and validation of the old ones.

Using lessons from experience

- Lessons learned from past accidental exposure should be incorporated into training. Radiation therapy facilities are encouraged to share their experiences of actual and potential safety incidents through participation in databases such as Radiation Oncology Safety Information System (ROSIS), Safety in Radiation Oncology (SAFRON), often referred to in this report.

Overcoming the lack of experience when introducing new technologies

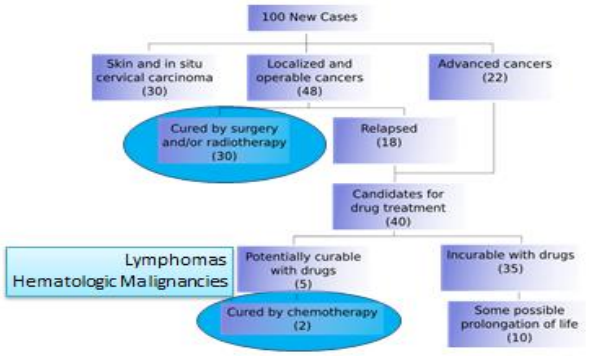
- Prior to the introduction of new techniques and technologies, there is little or no operational experience to share. Two complementary measures are recommended:
  - Prospective safety assessments should be undertaken in order to develop risk-informed and cost-effective quality assurance programmes.
  - Moderated electronic networks and panels of experts supported by professional bodies should be established in order to expedite knowledge sharing at the early phase of introducing a new technology.

Indications for Radiotherapy in Patients with Breast Carcinoma: Levels and Sources of Evidence

Outcome no.	Clinical scenario	Treatment indicated	Level of evidence*	Reference(s)	Proportion of all breast carcinoma cases
1	DCIS, breast-conserving surgery	Adjuvant radiotherapy	II	NHMRC National Breast Cancer Centre,[7] NCI,[13] NCCN,[14] Cancer Care Ontario Practice Guideline Initiative,[16] BC Cancer Agency,[23] The Steering Committee on Clinical Practice Guidelines for the Care and Treatment of Breast Cancer[22]	0.09
2	DCIS treated with mastectomy, local recurrence	Radical or adjuvant radiotherapy	III	NHMRC National Breast Cancer Centre,[8] NCI,[13] NCCN,[14] BC Cancer Agency,[23] Royal College of Radiologists CODN[25]	< 0.01
4	T1–T2, N0–N1, M0, breast-conserving surgery	Adjuvant radiotherapy	II	NHMRC National Breast Cancer Centre,[8] NCI,[13] NCCN,[14] Cancer Care Ontario Practice Guideline Initiative,[16] BC Cancer Agency,[23] Scottish Intercollegiate Guidelines Network[24]	0.62
5	T1–T2, N0–N1, M0 mastectomy, 0–3 positive lymph nodes, local recurrence	Radical or adjuvant radiotherapy	III	NHMRC National Breast Cancer Centre,[8] NCI,[13] NCCN,[14] BC Cancer Agency,[23] Royal College of Radiologists CODN[25]	0.01
12	T1–T2 N0–N1, M0, mastectomy, > 3 positive lymph nodes	Adjuvant radiotherapy	I	NBCC Radiation Oncology Advisory Group,[10] NCI,[13] NCCN,[14] NIH Consensus Development Panel[17]	0.03
13	T3–T4, any N, M0, and good/fair PS or any T, N2–N3, M0, and good/fair PS	Radiotherapy +/– systemic therapy, +/– surgery	III	NHMRC National Breast Cancer Centre,[8] Recht et al.,[12] NCCN,[14] BC Cancer Agency[23]	0.06
6, 17	Any T, any N, M1, painful bone metastases or T1–T2, N0–N1, M0, distant recurrence with painful bone metastases	Palliative radiotherapy	I	NHMRC National Breast Cancer Centre,[8] NCI,[13] ASCO,[15] Cancer Care Ontario Practice Guideline Initiative,[16] BC Cancer Agency,[23] Scottish Intercollegiate Guidelines Network[24]	0.01
9, 10	Any T, any N, M1, brain metastases or T1–T2, N0–N1, M0, distant recurrence with symptomatic brain metastases	Palliative radiotherapy +/– surgery	II	NHMRC National Breast Cancer Centre,[8] NCI,[13] Simpson et al.,[21] Loeffler et al.,[18] BC Cancer Agency,[23] Scottish Intercollegiate Guidelines Network[24]	< 0.01
Proportion of all patients with breast carcinoma for whom radiotherapy is indicated					0.83

Delaney, 2003

Let's look at the numbers .....:
How effective is radiotherapy, and how about chemotherapy?



Souhami et al

Children (0-20 y)

Radiotherapy Indication	Proj. No. Treatment
Palliative treatments	271
Brain (Medulloblastoma)	236
Head/neck	146
Bone and soft tissue sarcomas	40
Leukaemia	24
Kidney (including Wilm's tumour)	16
Lung/thorax	10
Colon and rectum	8
Lymphoma / Hodgkin	3

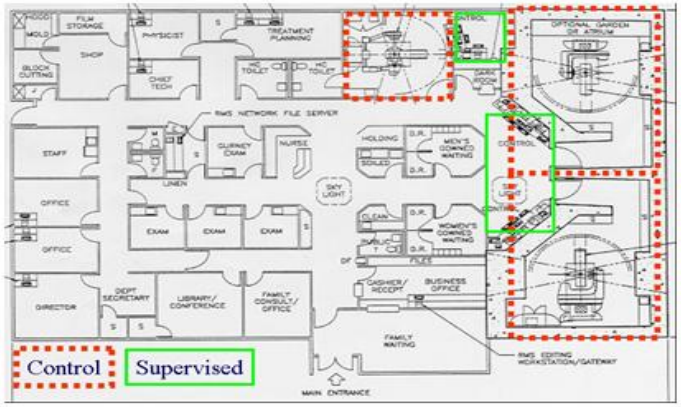
Utilization of radiotherapy still low for children: **<7% of all RT treatments**

Adequate Safety Infrastructure

- Design of a radiotherapy facility
  - **components** of a radiotherapy department
  - **design** criteria
- Shielding
  - **general** considerations
  - **external** beam radiotherapy
  - **brachytherapy**

Design Radiotherapy Facility

- Plan **size and workload** of the facility - be conservative, the design should be adequate for the next 20 year including room for **expansion**
- Consider **adjacent buildings**
- **Positioning the control room** and the equipment within so that staff have a **good view** of
  - the treatment room
  - access corridors
  - entrance to the treatment room
- **400 to 500 cases** per megavoltage unit per year

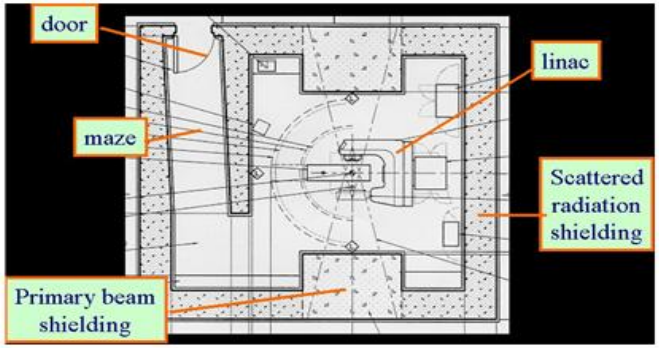


Controlled and supervised areas

- **Access restrictions**
- Require **warning signs**
- **Monitoring** of staff
- **Door interlocks** where appropriate
  - Protocol for closing the door and activating radiation
  - Door interlocks
- **Written procedures**

Treatment bunkers require lots of storage space for patient treatment aids...

Typical megavoltage room layout

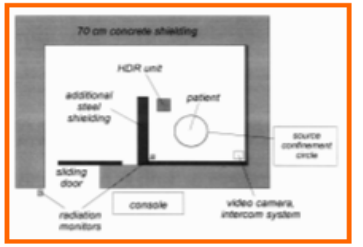


Design considerations: External Beam Radiotherapy

- **Placement** of the treatment unit
  - Primary beam direction
  - Operator location
  - Surrounding areas - should have low occupancy
- Patient and visitor **waiting areas** should be positioned so that patients are unlikely to enter treatment areas accidentally
- Patient **change areas** should be located so that the patient is unlikely to enter a treatment area accidentally
- Clear **signs are required** in areas leading to treatment units

Design considerations: Brachytherapy

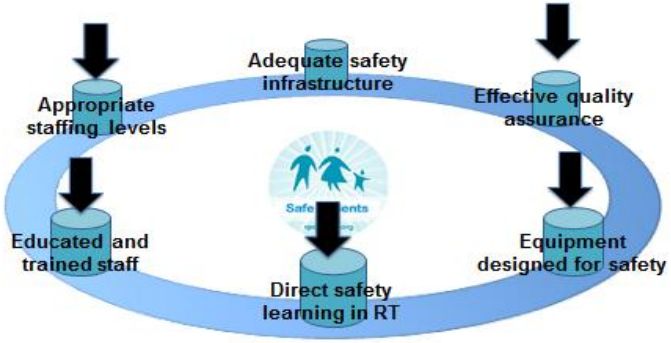
- A **secure storage area** for the source safe and **source transfer procedures** may be required
- The **design** of these rooms follow similar guidelines to those of accelerator rooms
- **Maze and door** must typically be included
- Similar **interlocks** to those used in accelerator rooms are required



A note of qualifications contractors … !!!

- There should also be a process in place which ensures that outside contractors on whom radiation protection may depend (*e.g.* service engineers) are appropriately trained and qualified. This is typically reflected in a **license of the contractor** of which a copy should be available in the radiotherapy facility.

Protection from Risks



Quality Assurance in Radiotherapy

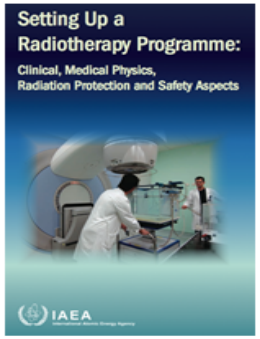
- **Quality assurance: planned and systematic actions** to ensure that a **RT facility consistently delivers high quality care leading to the best outcomes with the least amount of side effects**.
- Quality assurance has been replaced with *quality assessment* or *quality improvement* to emphasize the fact that it's a ***continuous, ongoing process.***
- **Quality control: procedures and techniques** used to monitor or test and maintain the components of the RT QI program.

Quality Assurance in Radiotherapy

- There are two main parts of quality assurance program in radiotherapy
  1. Periodic quality control **measurements and evaluation**
  2. Regular **preventive maintenance.**
- The three main areas for sources of inaccuracy in dose delivery can be identified as:
  1. **Physical dosimetry,** i.e. the commissioning and calibration of treatment machines and sources;
  2. **Treatment planning,** i.e. the delineation of target volume and critical structures, acquisition of patient specific factors and dose distribution calculations;
  3. **Patient treatment,** i.e. the set-up of the patient and the recording of the treatment and final verification of the accuracy of the delivered dose.

Quality Assurance in Radiotherapy

- The quality control programme in radiotherapy:
  - External beam treatments
  - Brachytherapy treatments
  - Measurement equipment
  - Clinical aspects of the treatments
- It should be noted that in many countries the specification, performance and quality control of teletherapy units may be subject to **government regulations**. If this is the case, these government regulations must be adhered to.



Quality Indicators

- **Quality Indicators:** measurement tools used to evaluate an RT services organization's performance.
  - ***Consultation and informed consent***
    - History and physical report in treatment record
    - Pathology report
    - Consent form signed by patient & radiation oncologist
  - ***Treatment planning***
    - QC program for equipment and treatment planning computer
    - Target volume indicated on target films
    - Setup information, diagrams, and photographs in treatment record
    - Calculation and graphic plans double checked.

Quality Indicators

- ***Treatment delivery***
  - QC program for equipment
  - Written and signed prescription
  - Approved treatment plan
  - Comparison of portal films with sim films
  - Weekly portal films signed by radiation oncologist and reviewed by radiation therapist.
- ***Documentation of Treatment delivery***
  - Adherence to prescription
  - Documentation of weekly physics review
  - Completeness of treatment record
  - Incidence/unusual occurrence reports
- ***Patient Outcomes***
  - Completion notes/summary and follow up notes filed in chart
  - Documentation of treatment outcomes

Equipment Design for Safety

- **Receipt, storage and disposal** of **radioactive sources**
- **Movement** of radiation sources and patients with sources inside the hospital
- **Safety devices**
- **Individual exposure monitoring**
- **Workplace monitoring**
- **Leak testing**
- **Maintenance and repair** of radiotherapy equipment, including obligatory notification to the qualified expert in radiotherapy physics before resumption of use (for a decision about whether beam measurements are necessary before resumption of treatments)

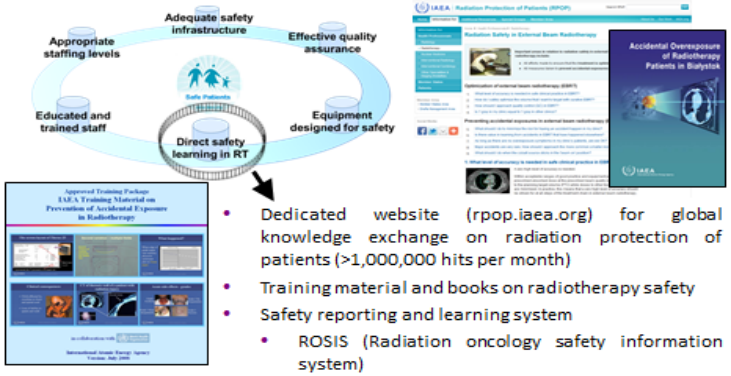
Equipment Design for Safety

- **Penumbra trimmer** may be employed to reduce penumbra width
- **Beam stopper**
- **R&V system** (record and verify system) records if the beam stops in the middle of treatment

Beam Stopper

Metal disk at the exit side:
- reduces primary beam shielding requirements
- may make set-up of patients more cumbersome

Protection from Risks

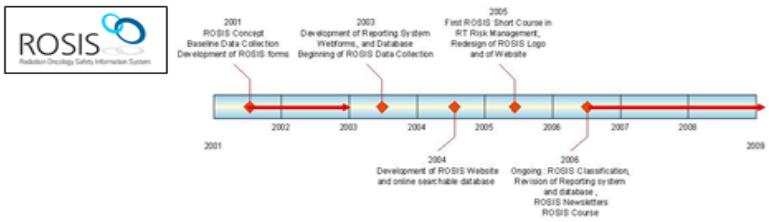


- Dedicated website (rpop.iaea.org) for global knowledge exchange on radiation protection of patients (>1,000,000 hits per month)
- Training material and books on radiotherapy safety
- Safety reporting and learning system
  - ROSIS (Radiation oncology safety information system)
  - SAFRON (Safety in Radiation Oncology)

Safety reporting and learning system

1. ROSIS (Radiation oncology safety information system)

- International web-based voluntary incident reporting system in radiotherapy
- Developed by a small group of health professionals in Europe
- Supported by ESTRO in the initial development stages



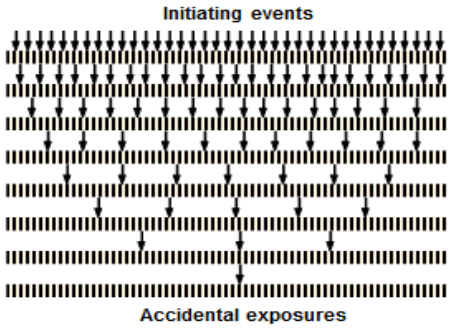
Safety reporting and learning system

2. SAFRON (Safety in radiation oncology)

- International web-based voluntary incident reporting system in radiotherapy
- Under development by the IAEA
- To be released (following pilot-study) later in 2012

Why Safety Reporting and Learning …. !!!

Safety in radiotherapy requires many safety-layers



Implementing lessons learned from reported events is only one of these layers

Personal Requirement for Clinical Radiation Therapy

Category	Staffing
Radiation oncologist-in-chief	One per programme.
Staff radiation oncologist	One additional for each 200–250 patients treated annually. No more than 25–30 patients under treatment by a single physician at any one time. Higher numbers of predominantly palliative patients can be managed.
Radiation physicist	One per centre for up to 400 patients annually. Additional in ratio of 1 per 400 patients treated annually.
Treatment planning staff: Dosimetrist or physics assistant	One per 300 patients treated annually
Treatment planning staff: RTT-Mould Room	One per 600 patients treated annually

Personal Requirement for Clinical Radiation Therapy

Category	Staffing
Radiation therapy technologist:	
Supervisor	One per centre
RTT	Two per megavoltage unit up to 25 patients treated daily; four per megavoltage unit up to 50 patients treated daily
RTT-Sim	Two for every 500 patients simulated annually
RTT-Br	As needed

Personal Requirement for Clinical Radiation Therapy

Category	Staffing
Nurse	One per centre for up to 300 patients treated annually and an additional one per 300 patients treated annually
Social worker	As needed to provide service
Dietician	As needed to provide service
Physiotherapist	As needed to provide service
Maintenance engineer or electronics technician	One per two megavoltage units or one megavoltage unit and a simulator if equipment serviced 'in-house'

Staff training and education

- All staff in radiotherapy must have **appropriate education** to perform their duties.
- The Regulatory Authority should encourage medical authorities, universities and professional organizations to design and **implement training programmes** in radiation protection for:
  - Radiation Protection Officer
  - Physician
  - Qualified Expert (Medical Physicist)
  - Radiotherapy Technician
  - Maintenance staff
- **Continuous education and training** in radiation protection shall be provided to meet with changes in equipment, instrumentation, practice, monitoring methods, recommendations and regulations

Staff training and education

- It is essential for all staff to have **regular updates** on radiation protection aspects
- Continuing education must be **documented**
- IAEA-TECDOC-1040 provides specific information on the role of the physician (3.2.2.1) and medical physicist (3.2.2.2) - the skills and competencies identified there determine the education and training requirements.

INVITED SPEAKER :

Dr. Susilo Widodo,

BATAN,

Indonesia

Environmental Radiation and Radioactivity Levels in Indonesia

Susilo Widodo, Dadong Iskandar
National Nuclear Energy Agency (BATAN)
Center for Technology of Radiation Safety and Metrology

Historical Background

Due to the increasing of the development and utilization of atomic energy for human welfare, Indonesia on December 5, 1958 established the Atomic Energy Council which is then refined into the National Atomic Energy Agency (BATAN) in1964.

One of the task is to search for any radioactive fallout and assess the levels in Indonesia teritorial due to nuclear test in Asia-Pacific region.

BATAN Terus Berkarya
55
Selamat, Aman, Sejahtera

Historical Background

- 1954: Establishment of State Committee for Radioactivity, concern over global fallout
- 1955: Establishment of UNSCEAR
- 1958: Establishment of Atomic Energy Council,
- 1964: Establishment of National Atomic Energy Agency (**BATAN**) – Act No. 31/1964
- 1965: Operation of Research Reactor Triga Mark II at Bandung : 250 kW
- 1970: NPT signed by Indonesian Government
- 1971: Operation of Upgraded Research Reactor Triga Mark II: 1 MW
- *1973: Indonesia Joined UNSCEAR*
- 1978: NPT ratified by Parliament
- 1979: Operation of Research Reactor Kartini at Yogyakarta: 100 kW
- *1986: Chernobyl Nuclear Accident*
- 1987: Operation of Multi Purpose Reactor RSG at Serpong: 30 MW
- 1997: Nuclear Energy Act No. 10/1997;
- 1998: Establishment of **BATAN** and **BAPETEN**,
- 1998: Economic and political crisis,
- 2000: : Operation of Upgraded Research Reactor Triga Mark II: 2 MW
- 2007: Act No. 17/2007 on National Long Term Development Planning;
- Introduction of NPP (2015-2019),
- *2011: Fukushima Daiichi NPP Accident*

Why radiation and radioactivity (R & R) ?

- Radiation ➔ deal with dose rate ➔ direct in situ measurements
- Radioactivity ➔ deal with natural and man-made radioactive material ➔ field sampling, measurements and analysis at laboratories

Environmental R & R Monitoring

- Radioactivity monitoring of fall out has been carried out continuesly since 1980 ➔ focused for Cs-137 and Sr-90 through food-chains samples: milk, grass and rain water (28 stations) ➔ negative results
- Base line data of environmental radioactivity in Indonesia ➔ some anomaly founded
- Development of mobile (carborne) monitoring system
- Atmospheric radioactivity monitoring stations are being developed ➔ CTBTO level, regional/global network

Environmental R & R Monitoring

In addition of fall-out monitoring, BATAN also undertakes measurement of radioactive concentration on imported and exported foodstuffs, such as milk products, spices, CPO, meat, etc. For example, after Fukushima Dai-ichi accident, BATAN measures imported foods from Japan those did not have radioactive certificate.

The measurement data of the imported foodstuff from Japan after Fukushima Dai-ichi accident has also been compiled by WHO and presented in the WHO Report on "Preliminary dose estimation from the nuclear accident after the 2011 Great East Japan earthquake and tsunami" (2012)

Mapping of Environmental R & R

Indonesia is the largest archipelago in the world, comprising five main islands – Java, Sumatra, Sulawesi, Kalimantan and Papua plus 30 archipelagoes totaling more than 17,500 islands (6,000 islands inhabited).

BATAN from 2005-2012 conducted mapping the environmental radiation and radioactivity level in Indonesia aiming to produce a map of the baseline data for development planning and other purposes.

The mapping is started with a big grid (60 km x 60 km), then if an anomaly founded, the measurement will be continued in smaller grid.

The smaller grid also applied for the vicinity of coal power plants.

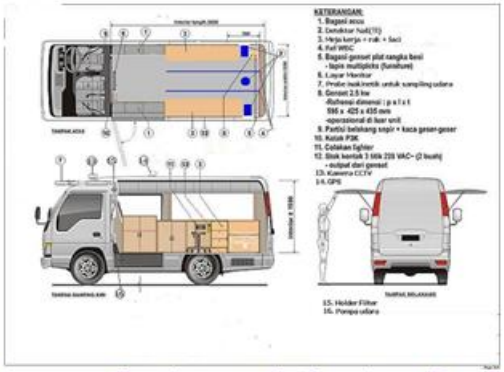
MAPPING METHODOLOGY

1. Systematic grid sampling has been used for sampling approach with grid size 60 km x 60 km for five main islands and 20 km x 20 km for other islands;
2. Environmental gamma dose rate performed continuously in the car during the trip in the grid and also in outside of car in soil sampling point;
3. Soil samples were taken at each grid with a depth of 0-5 cm and 5-20 cm ;
4. Radionuclide concentrations were measured by Gamma Spectrometer with HP Ge Detector.
5. The map were made using MapInfo Software v.10.5

Environmental R & R Monitoring

- Direct dose level measurements by using vehicle mounted Surveymeter and GPS
- Soil sampling, 60 x 60 km grid, 0 - 5 cm and 5 - 20 cm depth
- Direct meaurement at 1 m height

Environmental R & R Monitoring



a mobile system for integrated detection of radiation and environmental radioactivity with real time data

Environmental R & R Monitoring

a mobile system for integrated detection of radiation and environmental radioactivity with real time data

ENVIRONMENTAL GAMMA DOSE RATES (nSv/h) IN INDONESIA

	Range(nSv/h)	Geometric average (nSv/h)	Highest area
Gamma Dose Rates	10 - 8,000	56 ± 2	Mamuju,West Sulawesi

RADIUM-226 RADIONUCLIDE CONCENTRATION IN SOIL SURFACE IN INDONESIA

	Range(Bq/kg)	Geometric average (Bq/kg)	Highest area
Radium -226	5 - 7586	33 ± 3	Biak Island, Papua

THORIUM-232 RADIONUCLIDE CONCENTRATION IN SOIL SURFACE IN INDONESIA

	Range (Bq/kg)	Geometric average (Bq/kg)	Highest area
Thorium-232	3 - 3457	45 ± 4	Mamuju, West Sulawesi

POTASSIUM-40 RADIONUCLIDE CONCENTRATION IN SOIL SURFACE IN INDONESIA

	Range (Bq/kg)	Geometric average (Bq/kg)	Highest area
Potassium-40	3 - 1545	142 ± 3	Mamuju, West Sulawesi

Distribution of Population Dose in Indonesia

Absorbed dose rate in air (excluding cosmic-ray exposure)	Population(∞)
<20	49,000,000
20-29	44,000,000
30-39	47,000,000
40-49	21,000,000
50-59	11,000,000
60-69	4,000,000
70-79	3,000,000
80-89	2,000,000
90-99	5,000,000
100-199	7,000,000
200-299	5,000,000
>300	250,000

Mapping of Environmental R & R

Result summary:

- The average environmental gamma radiation in Indonesia is 45 nSv/h
- There are some areas with high natural radiation i.e. in Mamuju, Biak Island, Bangka Island, Karimun Island, and Tual Island. The highest radiation is in Mamuju, West Sulawesi.
- It is likely that the most high natural radiation areas are not located in granite rocks.

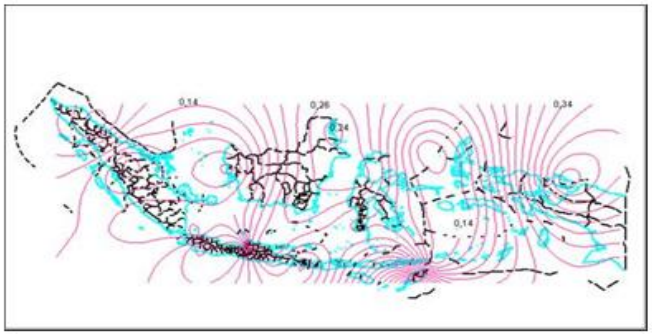
Environmental R & R Monitoring

A summary of the measurements results for Sr-90 and Cs-137 in rainfall from1982-2003, 28 stations

Gross beta	: undetected -	0.97 ± 0.14 Bg/m3
Sr-90	: un-detected -	0.81 ± 0.42 Bq/m3
Cs-137	: un-detected -	0.74 + 0.39 Bq/m3

Environmental R & R Monitoring

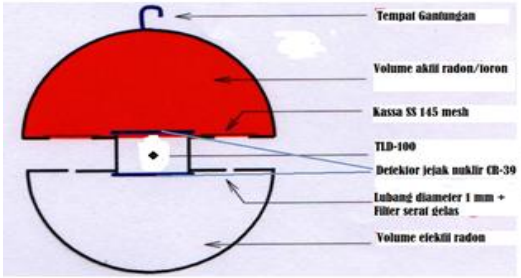
A summary of the measurements results for Sr-90 in rainfall from1982-2003



Further study at a high natural radiation of Mamuju, West Sulawesi:

Parameters measured:

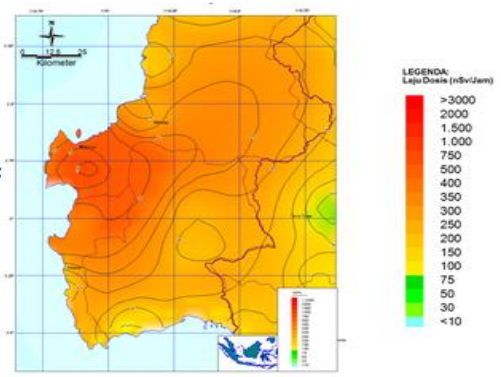
- Environmental gamma dose rates
- Radioactivity concentration in soil, vegetation, and drinking water
- Radon concentration in houses
- Radon exhalation rates from soil
- Preliminary study of chromosome aberation .

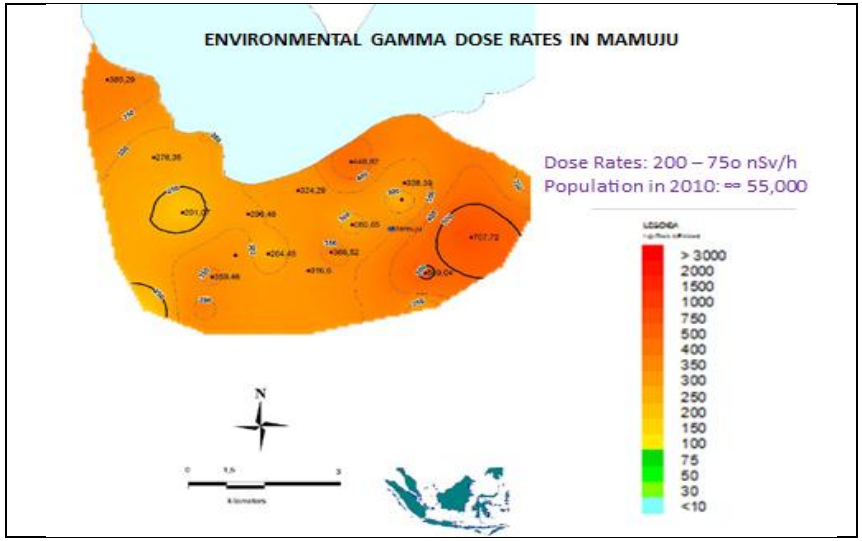


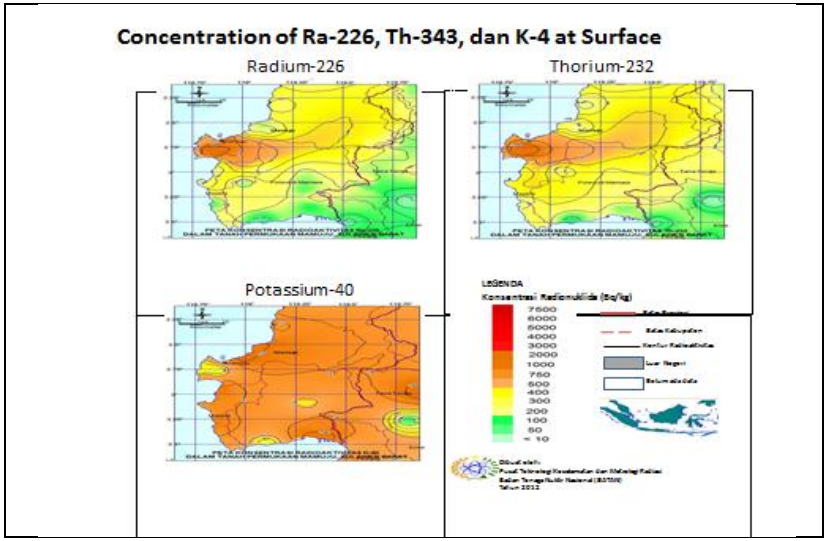
INDOOR RADON MEASUREMENT

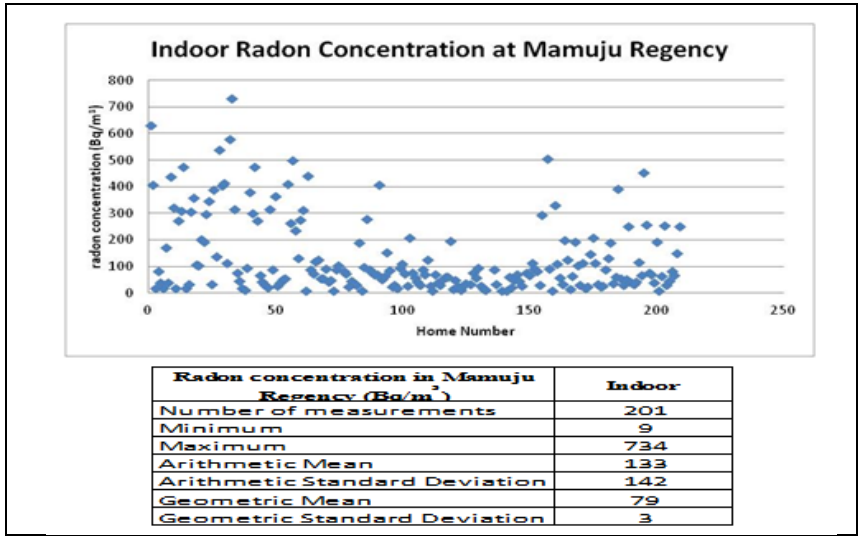
ENVIRONMENTAL GAMMA DOSE RATES IN MAMUJU

Dose rates:
200 – 8000 nSv/h
Population in 2010:
≈ 336,000

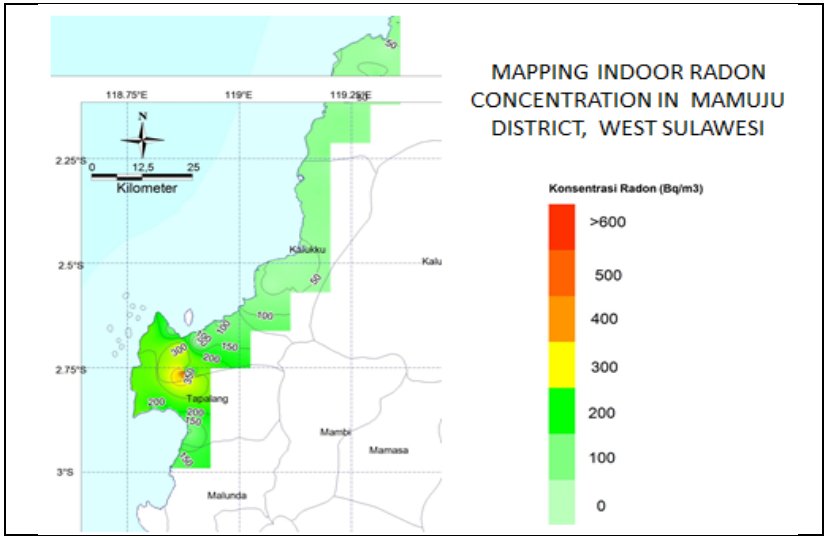








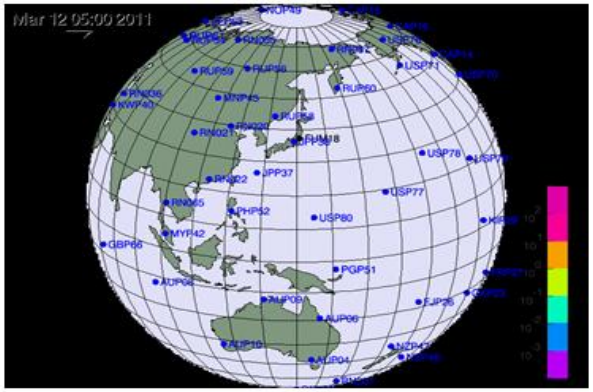
Radon concentration in Mamuju Regency (Bq/m³)	Indoor
Number of measurements	201
Minimum	9
Maximum	734
Arithmetic Mean	133
Arithmetic Standard Deviation	142
Geometric Mean	79
Geometric Standard Deviation	3



Response to Japan 11-3-2011 NPP Accident

- Coordination response meeting have been conducted with participants from various ministries and other governmental institutions, e.g. :
  - Ministry of Health,
  - Ministry of Agriculture,
  - Ministry of Trade,
  - Ministry of Industry,
  - Ministry of Marine and Fishery,
  - Ministry of Foreign Affairs,
  - BATAN (National Nuclear Energy Agency),
  - BAPETEN (Nuclear Energy Regulatory Agency)
  - BPOM (Food and Medicine Regulatory Agency),
  - BMKG (Meteorology, Climatology ,and Geophysics Agency), and
  - Custom Agency.

Atmospheric transport model (ATM) calculations : Fukushima Daiichi NPP



CTBTO - International Data Center

Response to Japan 11-3-2011 NPP Accident

- Screening and monitoring of possible radioactive contamination in civil aviation, people, foods, goods shipped from Japan to Indonesia after 15 March 2011,
- Conducted radioactivity sampling and analysis in environment (air, water, soil) in the northern region of Republic of Indonesia,
- Provided information of accident to the public through media.

Results:

- There was no radioactive contamination related to the accident detected in Indonesian territory,
- There was no radioactive contamination detected on airplanes and ships which arrived at Indonesian territory.
- There were five Indonesian citizens arrived by airplanes contaminated with extremly low level I-131 on their clothes and baggages.

Response to Japan 11-3-2011 NPP Accident

- Screening and monitoring points for radioactive contamination to people, air and water undertaken by BAPETEN for several days after accident

Response to Japan 11-3-2011 NPP Accident

BALI

- Screening and monitoring of radioactive contamination to people undertaken by BAPETEN for several days after accident at Bali airports

Response to Japan 11-3-2011 NPP Accident

BONTANG

SOEKARNO HATTA

- Screening and monitoring of radioactive contamination to people undertaken by BAPETEN for several days after accident at harbor and airports

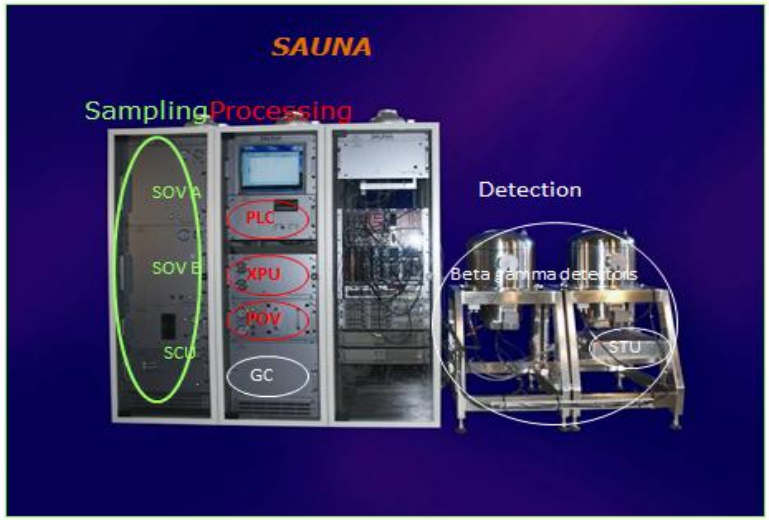
Contribution to UNSCEAR

Indonesia has contributed to the UNSCEAR global survey. The national data and two areas with high natural radiation (Bangka and Karimun Islands) has been reported for 2006 global survey.

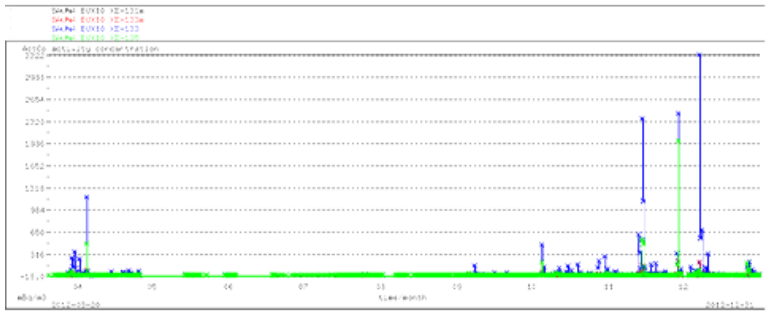
Concentrations in soil ($Bq\ kg^{-1}$)	^{40}K	^{226}Ra	^{232}Th
Average	197.19 ± 1.21	13.83 ± 0.83	12.34 ± 1.12
Range	75.12 – 522.71	7.29 – 54.22	2.40 – 58.10

Absorbed dose rates in air ($nGy\ h^{-1}$)		From cosmic radiation	From terrestrial radiation	Total
Outdoors	Average	30.33 ± 0.46	44.07 ± 44.29	74.40 ± 44.29
	Range	23.4 – 32.5	11.05 -235.08	4138 – 265.41
Indoors	Average	29.15 ± 3.98	-	-
	Range	2.98 – 3.06	-	-

Noble Gas monitoring using TXL-SAUNA II Installed in Pasar Jumat – Jakarta Selatan.



Typical Measurement Result of Noble Gas levels at Jakarta



Evaluated by Dr. Ulrich Stoehlker (CTBTO)

Atmospheric Monitoring with a High Volume Air/Particulate Sampling: under development

Environmental R & R Monitoring

Future Plan:

Sampling and measurements should address issues related to:

- Improvement of the standards and procedures
- Traceability of the measurement and methods,
- Sampling credibility, authenticity,
- ➔ adopt new technology, using GPS, photograph,
- On-line data transfer for *in situ* measurements
- Sample chain of custody,

Environmental R & R Monitoring

Future plan:

- BATAN: Installing CTBTO Level of atmospheric monitoring stations:
  Jakarta (2013) & Pontianak (2014)
- BAPETEN: nuclear installation (Jakarta, Bandung, Yogyakarta and trans-boundary

Thank you

Thank you

PANEL DISCUSSION :

Dr. Anhar Riza Antariksawan,

BATAN,

Indonesia

COORDINATION AND COLLABORATIVE WORKS

Anhar R. Antariksawan
National Nuclear Energy Agency (BATAN)

Why?

- Increasing the use of radioactive sources in many activities
- Increasing the demand on the safety from the use of radioactive sources
- Data collected on global levels and effects of ionizing radiation from ASEAN region needs to be increased
- Collaborative works on data collection and assessment of effects of ionizing radiations to provide scientific basis for radiation protection could help to improve the outcomes
- The outcomes could be beneficial to many countries and organization: UNSCEAR, IAEA, WHO etc

How?

- Use of existing network
  - Asian Nuclear Safety Network (ANSN)
    - Concern on nuclear safety
    - Incorporated into ANSN TG on Emergency Preparedness & Response
- Create new network
  - ASEAN Network on Radiation Protection and Safety (ASEAN-RadSafeNet)
  - Should be proposed through ASEAN-COST or MoU ASEAN-IAEA or others
- Principle: synergize and work alongside with other international/regional networks in the similar field

What?

- Field of networks:
  - Environmental Radiation and radioactivity
  - Worker and patient dose in diagnostic radiology and therapy
  - Medical response in nuclear/radiation emergency
  - Radiation dosimetry
- Ways and means:
  - Data exchange/data on-line
  - Intercomparison
  - Training programmes
  - Reference lab
  - Primary Standard Dosimetry Laboratory

THANK YOU
FOR YOUR ATTENTION

PANEL DISCUSSION :

Dr. Soehartati Gondhowiardjo,

Cipto Mangunkusumo Hospital

(RSCM)

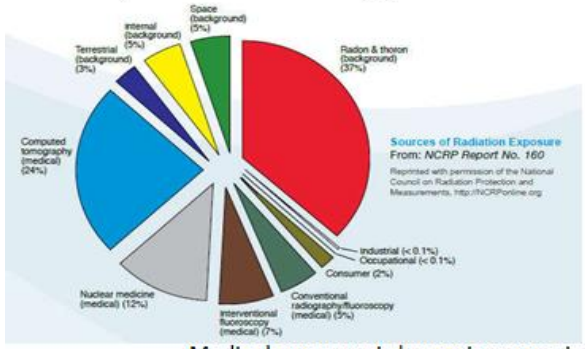
Indonesia

Gathering data on levels of medical exposure: Challenges?

Public / environment
Personals
PATIENTS

Prof. Soehartati, FKUI/RSCM

Annually worldwide, more than 3600 million x-ray exam performed, 3,7 million nuclear medicine done, and 7,5 million radiotherapy treatment - WHO



Medical exposure is becoming more important in the future: Are we ready to assess it?

Major Data Collection Needs on Medical Uses of Radiation

- Work load .....on going .... **Political issues**
- Outcomes (positive and negative)
  - Currently is of more interest ....**clinicians**
- Exposures
  - Intentional
  - Unintentional – incidents & accidents

Challenges for Indonesia

- The data: what to collect?
  - Reality:
    - Equipment specifications vary widely
    - Measuring instrument availability very low .... RD
    - A lot of data still in paper forms .... Electronic data base
    - Data entity not necessarily "patients" but can also "examinations", "procedures", "courses", "fractions", etc
    - Combining data is a challenge considering that we are working with paper!

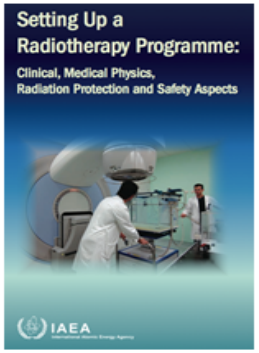
Challenges for Indonesia

- The data: what to collect?
  - Prospective data collection is necessary
  - Start small: what is "small"?
    - "easy" data on as many sources as possible? .. High coverage
    - "pilot" sources with as comprehensive / high quality data possible? ... low coverage (high quality)
    - Minimize "measured data"? "Pilot measurements?"
    - Combination of the above?

Challenges for Indonesia

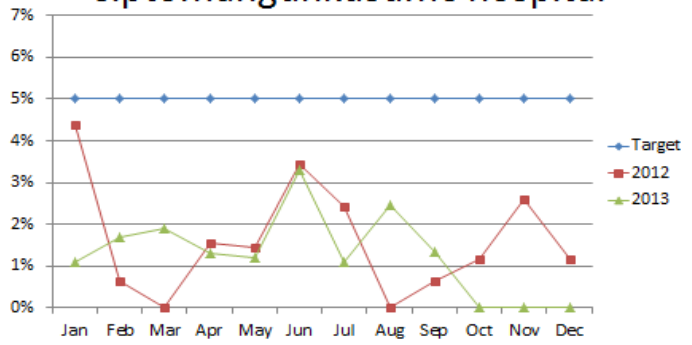
- Available databases: How complete are our databases?
  - Regulatory Agency (BAPETEN) Licensing Database
    - Depends on enforcement of regulations on licensing (it's improving)
    - Doesn't provide information on utilization
  - Hospital accreditation committee (KARS)
    - Might provide comprehensive information that include utilization and workload
    - "Jungle" of documents (in hardcopy!)
  - Society
    - Valuable in coordinating the data collection
    - Limited actual access to data (depends on voluntary reports)

For Radiotherapy: QUATRO



- A large amount of data is routinely collected and stored in radiotherapy departments
- In QUATRO missions, these are evaluated and checked
- → Valuable when in-depth information is necessary or confirmation is needed
- Problem: coverage (1-2 centers per year only)

What about unintentional exposure? (Incidents, Accidents) Ciptomangunkusumo hospital



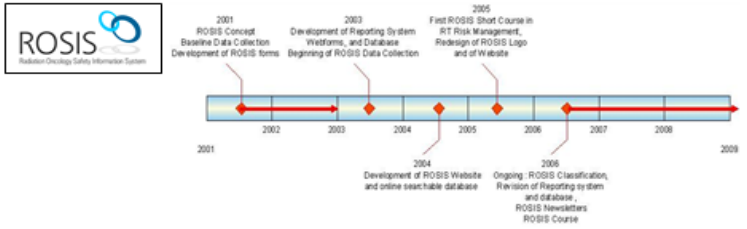
Protection from Risks

- Dedicated website (rpop.iaea.org) for global knowledge exchange on radiation protection of patients (>1,000,000 hits per month)
- Training material and books on radiotherapy safety
- Safety reporting and learning system
  - ROSIS (Radiation oncology safety information system)
  - SAFRON (Safety in Radiation Oncology)

Safety reporting and learning system

1. ROSIS (Radiation oncology safety information system)

- International web-based voluntary incident reporting system in radiotherapy
- Developed by a small group of health professionals in Europe
- Supported by ESTRO in the initial development stages



Safety reporting and learning system

2. SAFRON (Safety in radiation oncology)

- International web-based voluntary incident reporting system in radiotherapy
- Under development by the IAEA
- To be released (following pilot-study) later in 2012

Points:

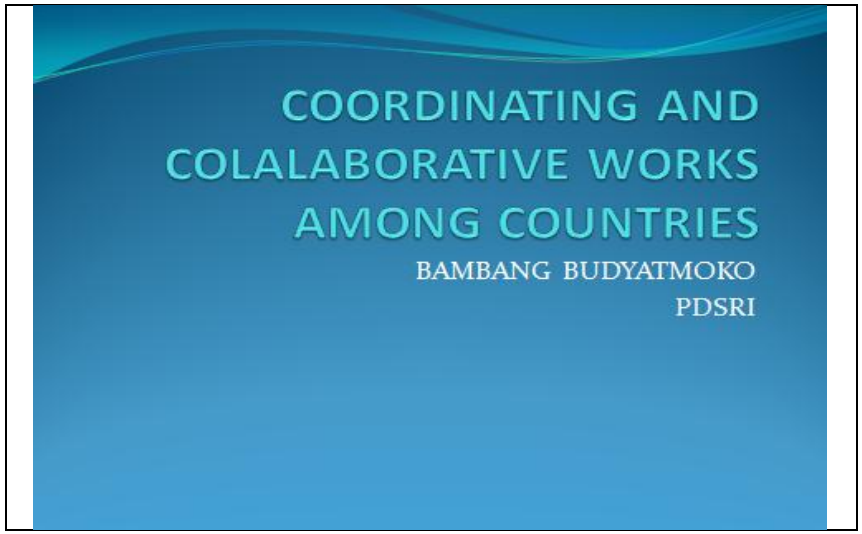
- Start small and realistic
- Set priorities: how to allocate funds?
  - Clinical research?
  - Incident reporting systems?
  - Exposure survey?
- Political issues .....
- Wealth of existing information: inter-institution sharing!

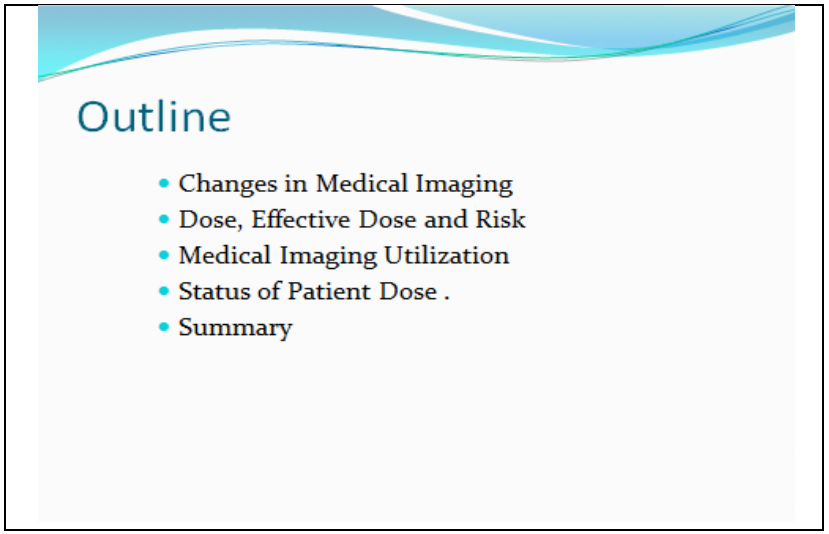
PANEL DISCUSSION :

Dr. Bambang Budyatmoko,

Indonesian Society of

Radiology (PDSRI)





Radiology

- The use of x-rays in medical imaging has shown rapid growth in the last 20 years
- This has been due to
  - Shorter imaging times
  - Increased indications for use
- This has increased concern over the associated radiation dose to patients

Benefit and Risk

- In all medical interventions there is a balance between benefit and risk
- In the vast majority of cases the benefit from diagnostic x-ray studies far outweighs any detriment

Patient Dose

- The risk from radiation is measured by Effective Dose,
- Background radiation is 2 mSv per year

ICRP (2008) Report 103

Procedure	Effective Dose mSv	Months of Natural Background
Very Low Dose		
Bone density scan	0.0002	0.001
Low Dose		
Skull series	0.05	0.3
Chest PA	0.05	0.3
Extremity	0.1	0.6
Thoracic Spine AP	0.5	3
Lumbar Spine AP	1	6
Mammography 2views	1	6
Abdomen AP	1	6
Intermediate Dose		
Pelvis AP	1.6	9.6
Head CT	2	12
Upper GI Series	2.5	30
IVP	3	24
Lower GI Series	6	24
Higher doses		
Chest CT	7	42
Abdomen CT	9	54
Pelvis CT	9	54
Cardiac Angiogram	8	48
Natural Background	2	12

Patient Doses

Procedure	Effective Dose mSv
No Dose	
MRI, US	ZERO
Low Dose	
CXR Extremities	<1
Intermediate Dose	
IVP, lumbar spine, abdomen bone scan, CT head and neck	1 - 5
Higher doses	
Chest or Abdomen CT Nuclear cardiogram Cardiac angiogram	5 - 10
Natural Background	2

Guidelines

The chief causes of the wasteful use of radiology are:

- Repeating investigations which have already been done
- Investigating too often
- Doing the wrong investigation
- Failing to provide appropriate clinical information and questions

Patient Dose (in Indonesia)

- We have not continuously monitored patient dose
- There is no regulation from the hospital ?
- Ministry of Health ?
- Bapeten ?

CT Survey 2004 From 18 Hospital

Study	Range mSv	Mean Dose mSv
Head	2 – 5	2.8
Chest	3 – 27	9.0
Abdomen	4 – 27	10.2
Pelvis	4 – 16	9.0
Abdo-Pelvis	7 – 32	16.5

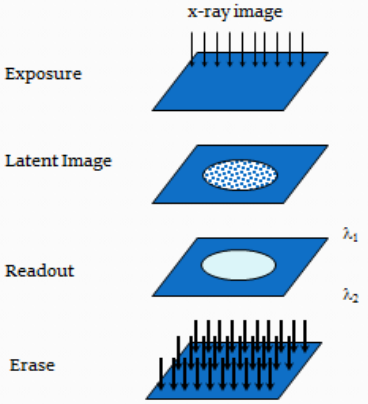
Aldrich et al CARJ 2006;57:281

Comparison of Mean Doses (mSv)

CT Study	EU 1999	US 2000	Germany 2002	UK 2003	BC 2004
Head	2.0	2.0	2.8	2.1	2.8
Chest	8.8	9.1	5.7	9.9	9.0
Abdomen	7.8	8.3	-	7.1	10.2
Pelvis	7.9	5.4	7.2	-	9.0
Abdomen -Pelvis	-	12.1	14.4	9.5	16.5

Computed Radiography(CR)

Cassette - based system using a storage phosphor instead of film-screens



Summary – CR Survey

Radiographic Study	Relative Dose
AP Abdomen	4 – 14
PA Chest	4 – 22

Diagnostic Reference Levels

Exam	Dose Area Product Gy cm^2	DAP to E Conversion Factor mSv/Gy cm^2
Skull AP	2	0.03
Chest PA	0.11	0.12
Abdomen AP	2.6	0.2
Pelvis AP	2.1	0.2
Barium Meal	14	0.2
IVP	14	0.1
Coronary Angiography	29	0.15

Skin Doses

- High doses to the skin of the patient can cause erythema and even hair loss
- This only occurs during long interventional angiographic procedures
- The threshold for these effects is about 2000 mGy to the skin

Angiography Skin Doses Published Data

Procedure	Diagnostic mGy	Therapeutic mGy
Cerebral Angiography	1200	1310
Carotid Angiography	215	154
Thoracic Angiography	260	-
Hepatic Angiography	360	540
Renal	620	660
Lower Extremity	68	146
Upper Extremity	73	150
Coronary catheterization	410	
PTCA		760
PTCA with stenting		1800

1. Bor et al BJR 2004:77:315
2. McParland et al BJR 1998:71:175
3. Van de Putte et al BJR 2000:73:504

Patient Dose Monitoring

It is now *technically* possible to estimate patient dose from most x-ray systems

Equipment	Dose Indicator	Status
CT	Dose-length product DLP	Available on all CTs
Radiographic/ Fluoroscopic/ Angiographic	Dose-area product DAP	Available on all new units- can be retrofitted
CR	Exposure Index (EI) of detector	All CR

Both DLP and DAP values are directly related to patient dose

Summary - Regulation/Education

- Regulation and Guidelines should be made
- Ministry of Health, Bapeten ?
- More information needs to be provided to users of diagnostic imaging on patient dose
- Training course for physicians who use fluoroscopy

Summary - Referring Physicians

- Avoid repeating investigations which have already been done
- Choose the appropriate investigation
- Provide appropriate clinical information and questions

Summary - Radiologists, Technologists

- Review patient doses
  - Use Diagnostic Reference Levels for comparison
  - Optimize techniques based on this information
  - Relatively easy to reduce higher doses

PANEL DISCUSSION :

Dr. Carl Magnus Larsson,

ARPANSA,

Australia

Australian Government
Australian Radiation Protection and Nuclear Safety Agency

Communicating radiation risks

Increasing awareness and improving understanding among stakeholders of the levels of ionising radiation and the related health and environmental effects

Panel discussion

International Conference on Sources, Effects and Risks of Ionizing Radiation

Sanur – Bali, 10 October 2013

Australian Government
Australian Radiation Protection and Nuclear Safety Agency

Understanding radiation risk

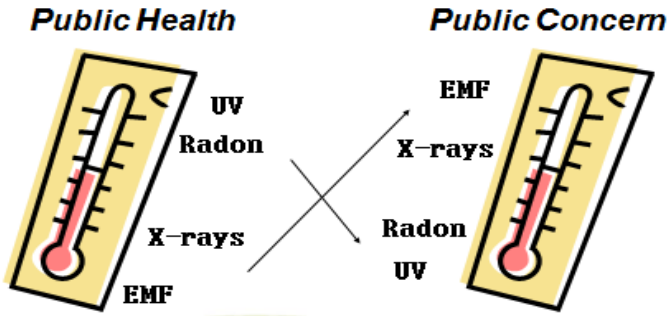
- LNT (no exposure without risk……)
- Epidemiology (no discernible effect …..)
- Terminology
  - absorbed dose, equivalent dose, effective dose, committed dose, collective dose, averted dose….. and more
  - curie, becquerel, gray, sievert (milli-, micro-, man-), rem, rad……and more
  - deterministic, tissue reaction, acute, late, stochastic, cancer, hereditary …..and more
  - dose limits, dose constraints, activity limits, reporting levels, operational intervention levels…..and more

?

2

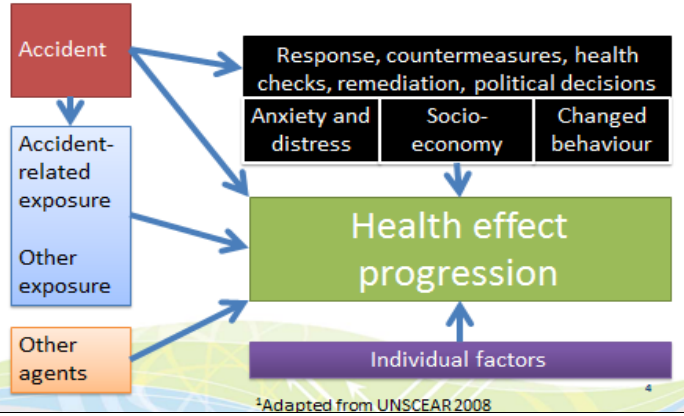
Australian Government
Australian Radiation Protection and Nuclear Safety Agency

Risk: Big or Small?



Australian Government
Australian Radiation Protection and Nuclear Safety Agency

Health risks from accidents[1]



[1]Adapted from UNSCEAR 2008

4

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Australian Radiation Protection and Nuclear Safety Agency

Crisis communication

5

Australian Government
Australian Radiation Protection and Nuclear Safety Agency

Challenges

- How to be credible
  - authenticity
  - truthful
  - stick to the agreed process
- How to talk about radiation
  - understandably; and in
  - relation to other relevant risks; so that
  - individuals and society can take their own informed decisions

6

Present Status of ^{137}Cs Concentration at Bangka Coastal, Indonesia

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Abstract. **Bangka island is a site of candidate of the first Indonesia nuclear power plant (NPP). In pre operational NPP, the monitoring of ^{137}Cs is necessary to be performed for practices to establish 'baseline' environmental radionuclides concentrations. On the other hand Indonesia has been following IAEA RAS/7/016 Project on Establishing a Benchmark for Assessing the Radiological Impact of Nuclear Power Activities on the Marine Environment in the Asia-Pacific region. Data of ^{137}Cs on Indonesia marine environment must be obtained to evaluate the extent and the possible impact. Seawater samples and sediment sample were collected from Bangka Coastal areas. The marine biota samples were bought directly from fisherman at local markets. After the samples were processed chemically according to standard procedures, the ^{137}Cs concentration was determined by gamma ray spectrometry using Canberra GX2018, Canberra GC2020 and Ortex GMX 25P4-76. Result of analysis showed that the concentration of ^{137}Cs in sediments, seawater and marine biotas were in the range from < MDA to 2.33 Bq.kg^{-1}, 0.49 to 0.66 Bq.m^{-3} and 4.02 to 109.75 mBq.Kg^{-1}, respectively.**

Keywords: NPP, Baseline of ^{137}Cs concentration, Bangka Island, Indonesia

Introduction

The radioecological marine monitoring in Indonesia is being done for a number of reasons. One objective is to define the levels of radioactivity in the marine environment as a baseline before the first NPP starts to operation. Indonesia had a plan to build the first NPP and one candidate of the locations chosen is Bangka Island that included in the Province of Bangka Belitung. The province of Bangka Belitung is located between 104°50' - 109°30' East Longitude and 0°50' - 4°10' South Latitude. The Bangka Strait separates Sumatra and Bangka, and the Gaspar Strait separates Bangka and Belitung. The South China Sea is to the north, the Java Sea is to the south, and the province is separated from Borneo in the east by the Karimata Strait.

Radioisotope of ^{137}Cs is a fission product with a moderately long half-life (30 years), and usually present as simple cations with high solubility and mobility in marine environments, depending particularly on the sorption of Cs to sediment surfaces (Borretzen *et al* 2002, Gaur, 2006, Ahmadi *et al* , 2011). In this study, baseline levels of ^{137}Cs were established in seawater, surface sediments and biota (fish, crustaceans and mollusks) along the Bangka island coast, with special emphasis on the area close to the candidate location of NPP, during the period of 2011-2012. The concentration level of ^{137}Cs had been reported in Malaysia coastal area (Yii *et al* 2007). This data could not be used as the base line data for site of NPP in Indonesia. It is caused by the fact that the concentrations of anthropogenic radionuclides in the marines are generally vary from region to region, according to the location and magnitude of the different sources of contamination (Figueira *et al* 2006).

The aim of this study was to collect baseline data for anticipation of the possible shift data that result from input of some radionuclides pollution in marine base source of pollution before operation of NPP. On other hand Indonesia has been followed IAEA RAS/7/016 Project on Establishing a Benchmark for Assessing the Radiological Impact of Nuclear Power Activities on the Marine Environment in the Asia-Pacific region. Data of ^{137}Cs on Indonesia marine environment must be obtained to evaluate the extent and the possible impact. Unfortunately there were no any data of artificial radionuclides concentration in coastal environment before Fukushima accident. These activities contained the measurement of ambient dose rate, sampling, preparation and analysis and data interpretations. This paper presents analysis of ^{137}Cs in seawater, sediment and biota from Bangka Coastal in Indonesia.

Materials and Methods

Seawater samples were collected from Bangka Coastal areas with some points of collection is shown in Figure 1. One hundred liter samples were filtered and acidified to pH 3 and 11 grams of Ammonium Molybdophosphate (AMP) was added. The AMP added sample was filtered through a 5-cm diameter filter paper and kept on a plastic petri dish. The ^{137}Cs activities were determined by gamma spectrometry. The chemical procedures used for the seawater samples were based on Inoue *et al.* (2011). The ^{137}Cs and ^{226}Ra concentration were determined by gamma-ray

spectrometry. In this experiment it was used 3 HPGe detector with counting efficiency of 20 - 25% and FWHM of 1.8 keV. The type of these gamma spectrometers were Canberra GX2018, Canberra GC2020 and Ortex GMX 25P4-76. The counting efficiency was calibrated by using suitable substrates and geometries.

Nine types of marine biota samples were bought directly from fishermen at local markets. About 5.0 kg of each biotas muscle were ashed at 400 °C for more than 24 h and transferred to a 7-cm polyethylene pot. The ^{137}Cs activity was determined as above. About 5 liters wet sediment sample were taken using a grab sampler and, after dried, 400-500 grams were weighed on a 7 cm polyethylene pot. After 30 days, to allow a simultaneous ^{226}Ra determination, the samples were analyzed by gamma spectrometry as above.

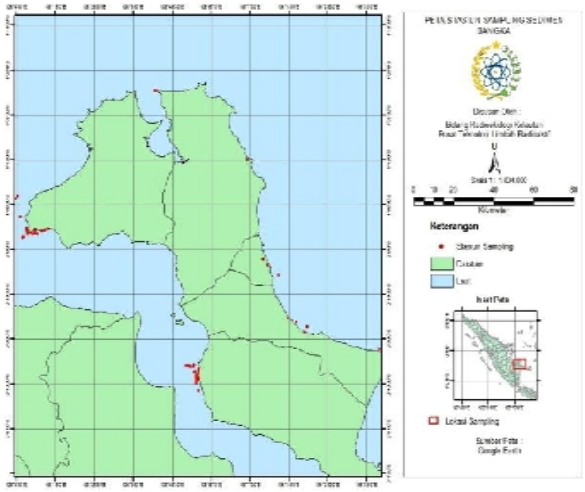

Figure 1. Sampling location at Bangka Island

Results and Discussion

The concentrations of ^{137}Cs in coastal sediments, water, and local biota were presented in Figure 2, Table 1 and Table 2.

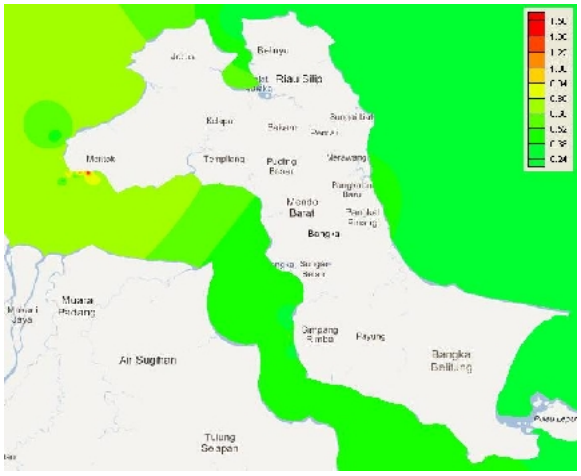

Figure.2 The sediments concentration profile of ^{137}Cs (Bq.Kg^{-1})

Table 1. The concentration of ^{137}Cs in seawater.

Location	^{137}Cs (Bq.m^{-3})
100° 27.400' E 01° 15.700' S	0.12 ± 0.02
100° 25.900' E 01° 14.850' S	0.10 ± 0.02
105° 06,334' E 02° 00.508' S	0.75 ± 0,05
105° 06,425' E 02° 02.402' S	0.84 ± 0,05
105° 07,040' E 01° 57.369' S	0.88 ± 0,03
105° 07,144' E 02° 00.156' S	0.75 ± 0,03
105° 07,696' E 02° 05.159' S	0.97 ± 0,05
105° 48.950’ E 02° 36.450’ S	0.05 ± 0.01
105° 51.550’ E 02° 36.250’ S	0.03 ±0.009
105° 53.800' E 02° 36.900' S	0.05 ±0.01
107° 37.833' E 02° 44.776' S	0.08 ±0.01
107° 40.160’ E 02° 33.500’ S	0.05±0.01

Table 2. The concentration of ^{137}Cs in marine biota.

Biota	^{137}Cs (mBq.Kg^{-1})
Arius thalassinus	14.15 ± 1.51
Lates calcarifer	62.09 ± 7.44
Scomberomorus commerson	109.75 ± 15.32
Scylla sp	4.02 ± 0.37
Plotosus lineatus	36.53 ± 4.05
Penaeus merguiensis	6.16 ± 0.59
Caesio erythrogaster	8.95 ± 9.30
Anadara granosa	10.65 ± 1.12
Euristhmus microceps	4.68 ± 0.60

All samples were taken between April 11 to December 2012 from more than 60 sampling locations (sediments) and 12 sampling locations (water). The concentration of ^{137}Cs in sediments and seawater were ranged from <MDA to 2.33 Bq.kg^{-1} and 0.49 to 0.66 Bq.m^{-3}, respectively. Meanwhile, its concentrations in marine biota were ranged from 4.02 to109.75 mBq.Kg^{-1}. The ^{134}Cs concentration in all samples were could not be detected. On other hand, there were no data as comparison of ^{137}Cs in this area. The highest concentration of ^{137}Cs in common local marine biota was in mackerel *(Scomberomorus commerson*) that was 109.75 mBq.Kg^{-1}. The differences of ^{137}Cs concentration in marine biota was caused by radiocaesium accumulation by marine biotas that was a complex dynamic process, which is determined by both environmental and physiological factors such as contamination of feedstuffs, feeding intensity, position in the food chain, etc (Kryshev, 2000).

There were studies conducted in some Asia regions and other areas before and after Fukushima accident to obtain the information regarding the ^{137}Cs concentration. Yii *et al* (2007) reported that in Malaysia coastal area, the concentrations of ^{137}Cs in sediments were in the range from <MDA to 7.92

Bq.Kg^{-1}. Duran et al. (2004) reported the concentration of ^{137}Cs in sediments and sea water in ASPAMARD were in the range from 0.08 to 23.4 Bq.Kg^{-1} dry and 0.2 to 8.2 Bq.m^{-3}, respectively. Kim et al. (2006) reported the concentration ^{137}Cs in the sediment in the East Sea near the southeastern part of the Korean Peninsula samples was in the range of <MDA to 7.19 Bq.Kg^{-1} dry. Moreover Kim et al. (2012) reported the concentrations ^{137}Cs in seawater and fish in Korean Peninsula were in the range of <1.16 to 2.55 mBq.Kg^{-1} and 0.09 to 0.25 Bq.Kg^{-1} fresh, respectively. Huh *et al* (2004) reported the average concentration of ^{137}Cs in seawater, algae and sediments from Northern Taiwan were 0.0024, 0.09, 0.21 and 0.32 Bq.Kg^{-1}, respectively.

Godoy et al. (2003) reported the ^{137}Cs concentration in water, sediments and fishes were 0.85 to 4.0 Bq.m^{3}, 0,28 to 3.8 Bq.Kg^{-1} and 0.01 to 1.6 Bq.Kg^{-1}. Pittauerová et al. (2011) reported the concentration of ^{137}Cs in river sediments and grass were 1.73 to 2.6 Bg.Kg^{-1} and 0.18 to 1.9 Bg.Kg^{-1}, respectively. The author concluded that the major portion of ^{137}Cs must originated from atmospheric bomb test and Chernobyl accident contributions. Hong et al. (2002) reported that concentration ^{137}Cs in fish (*atka makarel* and *flounder*) and shrimp were 0.33 to 10 mBq/Kg^{-1}. On other hand its concentration in walleye pollock (*Theragra chalcogramma*) was in the range of 0.13 to 0.18 Bq.Kg^{-1}. Antovic et al. (2011) reported the concentration of ^{137}Cs in whole individuals of the mullet species *C. labrosus* from the South Adriatic Sea were <MDA to 1.61 Bq.Kg^{-1}. Its concentration in sea water was 0.06 Bq.l^{-1}. Gasco et al. (2002) reported the concentration of ^{137}Cs in sea water that from the incoming Atlantic flow and in the outpouring Mediterranean water were 2.52 ± 0.28 Bq.m-3 and 2.14 ± 0.52 Bq m^{-3}, respectively. Inoue et al. (2011) reported the concentration ^{137}Cs in seawater samples that were collected at 10 sites along the coastline of the Northern Sanriku and Tsugaru Strait, 250 -450 km north of the Fukushima Dai-ichi NPP in May to June 2011 were 1.9 to 3.9 mBq.l^{-1}. Barsanti et al. (2012) reported the concentration of ^{137}Cs water sample and mussel was below MDA.

All measurements of samples above showed that concentration of ^{137}Cs was comparable to data in Asian and other regions before Fukushima accident. It was happened because the Fukushima fallout moved to north-eastern from Japan over the Pacific Ocean toward the Arctic Ocean, and then crossing the Atlantic Ocean over Iceland and spreading over the European continent (IRSN, 2011).

Conclusions

Results of analysis showed that the baseline concentrations of ^{137}Cs in sediments, seawater and marine biotas were in the range from <MDA to 2.33 Bq.kg^{-1}, 0.49 to 0.66 Bq.m^{-3} and 4.02 to109.75 mBq.Kg^{-1}, respectively. These concentrations of ^{137}Cs were comparable to other data collected in Asian and some other regions before Fukushima accident.

Acknowledgements

This work was supported by the Competitive Research Incentive Programme (SINAS) - The Indonesia Ministry of Research and Technology.

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DISCUSSION

1. Dr. Ohtsura Niwa, Fukushima Medical University, Japan :

   Question : Did you detect Cs-134 during marine monitoring?

   Answer : We never found spectrum Cs-134 during analysis marine sample with gamma spectrometer. We know that Cs-134 is a clue to estimate Fukushima-derived radiocesium and we concluded that there was no impact Fukushima accident to Indonesia marine environment.

2. Dr. Lebai Juri, MOSTI Malaysia :

   Question: Malaysia also conduct the marine monitoring after Fukushima accident. Did your data is comparable with Malaysia data?

   Answer : Yes our data is comparable with Malaysia data.

Application of Radiometric Mapping for Naturally Occurring Radioactive Materials (NORM) Assessment in Mamuju, West Sulawesi*)

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Abstract. **Mamuju has been known to have high radiation dose rate which is comes from NORM in rock and soil. Major concern is due to its location which is near inhabitant settlement area. Preliminary research has been done by environmental team which is limited to main access road only, while some remote area has left untouched. The purpose of the research is to delineate the location and distribution of thorium and uranium anomaly in Mamuju, and also to provide adequate information regarding the anomaly and high dose rate area to decision makers and stakeholders in neither local nor central government. Method applied is radiometric mapping using spectrometer RS-125 with NaI(Tl) detector in the area of interest Geological Formation of Adang Volcanic, which is more than 800 square km in size. The radiometric mapping method is widely used in uranium/thorium exploration, and now has been added with the measurement of radiation dose rate which is appropriate to environmental surveys. The mapping has been successfully delineates the area of NORM or the area with thorium and uranium anomaly. Thorium and uranium anomaly identified related with multi-geological- process resulting the increase of grade into several fold from its original state.**

Keywords: NORM, Mamuju, thorium, uranium, radiometric

*) The complete paper will be published in Atom Indonesia Journal.

Comparison of EPA 900.0 with SNI ISO 9696:2009 and SNI ISO 9697:2009 Methods in the Determination of Total Alpha and Beta Radioactivities in Water

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Abstract. **Measurement of gross alpha and gross beta is a simple method used for the determination of preliminary test of alpha and beta radioactivities in water for public health and nuclear emergencies purposes. Measurement of gross alpha and beta can be conducted by using EPA 900.0 method and SNI ISO 9696: 2009 / SNI ISO 9697: 2009 and the purpose of this research was to compare both methods. After preparation of the samples, for EPA 900.0 method the obtained sample with maximum total dissolved solid (TDS) of 500 mg/L and maximum measurement residues of 0.100 g for alpha and 0.200 for beta. For method of SNI ISO 9696:2009 and SNI ISO 9697:2009 the obtained sample with measurement residue between 0.1734 g and 1,0000 g or TDS between 65.41 mg/L and 350.44 mg/L for 1 L volume of water sample. EPA 900.0 method gave the recovery of 59.31 % for alpha counting (^{241}Am standard), 92.90 % for beta counting (^{90}Sr/^{90}Y standard), and with mix standard (^{241}Am+^{90}Sr/^{90}Y) it were 41.87 % for alpha counting and 92.26 % for beta counting. Whereas SNI ISO 9696:2009 and SNI ISO 9697:2009 methods gave the recovery of 10.22 % for alpha counting (^{241}Am standard), 72.46 % for beta counting (^{90}Sr/^{90}Y standard), and with mix standard (^{241}Am+^{90}Sr/^{90}Y) it were 10.94 % for alpha counting and 81.78 % for beta counting. Based on times required to analysis, EPA 900.0 method needed ± 6 h and SNI ISO 9696:2009 and SNI ISO 9697:2009 methods needed ± 9 h. Based on materials and tools, EPA method was simpler and for overall EPA 900.0 method was more effective dan efficient.**

Key words*: Gross alpha and beta activity, Methods of SNI ISO 9696:2009, SNI ISO 9697:2009, recovery.*

Introduction

Determination of radioactivity in environmental samples is extremly important due to related to human health and nuclear emergencies. Determination of gross alpha and beta radioactivities is a simple, inexpensive and fast method to detect radioactivity level in drinking water, the environment samples or liquid wastes (Semkov et al., 2004; Zapata-Garcia et al., 2009; Martin Sanchez et al., 2009; Wisser et al., 2006). Some methods that can be used for analysis of gross alpha and beta are EPA 900: 1980, ISO 9696:2007, ISO 9697:2007, and ASTM D1890-05: 2005. These methods are used to change the water sample to solid state throught several stages, typically used in the evaporation and subsequently placed onto planset stainless steel and measured the gross alpha and beta activities by gas proportional counter (Montana et al., 2012).

In this study, method of EPA 900.0 2009 was used to measure gross alpha and beta radioactivities in drinking water (EPA, 2009) and the method of measurement ISO 9696:2009 for Total Alpha Activity in Freshwater - Thick source method and ISO 9697 : 2009 for Total Beta Activity Measurements in the Freshwater - Thick source method were used (BSN 2009a; BSN, 2009b). Analysis of gross alpha and beta radioactivities is done quantitatively by using a Low Background Counter (LBC). This method can meet the requirements for the selection of samples and can be used for the determination of alpha and beta activity in a sample without distinguishing the origin of the radionuclides, so that all measurable alpha and beta activities are from all radionuclides present in the sample mixture. This analysis method has the advantage that it can be used for very many shots, it can be used to compare the levels of gross alpha and beta activity in freshwater and can choose which footage can be further analyzed (Iswantoro, 2008; Wahyuningsih et al., 2011).

The purpose of this study was to compare the EPA 900.0: 2009 method with SNI ISO 9696:2009 and SNI ISO 9697:2009 methods in determining the content of total gross alpha and beta radioactivities in water.

Materials and Methods

Analysis with EPA 900.0 method was done by evaporating 500 mL of water samples that had been added 2 mL of concentrated nitric acid until nearly dry. The sample was transferred directly onto the stainless steel planset and evaporated under an IR lamp until dryness and leveled using methanol. Calibration sources were obtained by making a standard of ^{241}Am for alpha with 14.16 Bq activity, a standard of ^{90}Sr/^{90}Y beta for beta with 10.00 Bq activity and mixed ^{90}Sr/^{90}Y +^{241}Am standard with varied weight in planset.

Analysis with SNI ISO 9696:2009 and SNI

ISO 9697:2009 methods was performed by evaporating 1 L of water sample to a 50 mL volume, and then put on a porcelain dish and then was added 1 mL of sulfuric acid (H_2SO_4), evaporated to dryness under an IR lamp and burned at a temperature of 350°C in a furnace. Calibration sources used were same as above. The radioactivities in samples were measured by using LBC Eclipse S550 LB with the counting time of 1 hour and data processing was carried out as standard procedure. Chart of procedures are presented in Figure 1 and Figure 2.

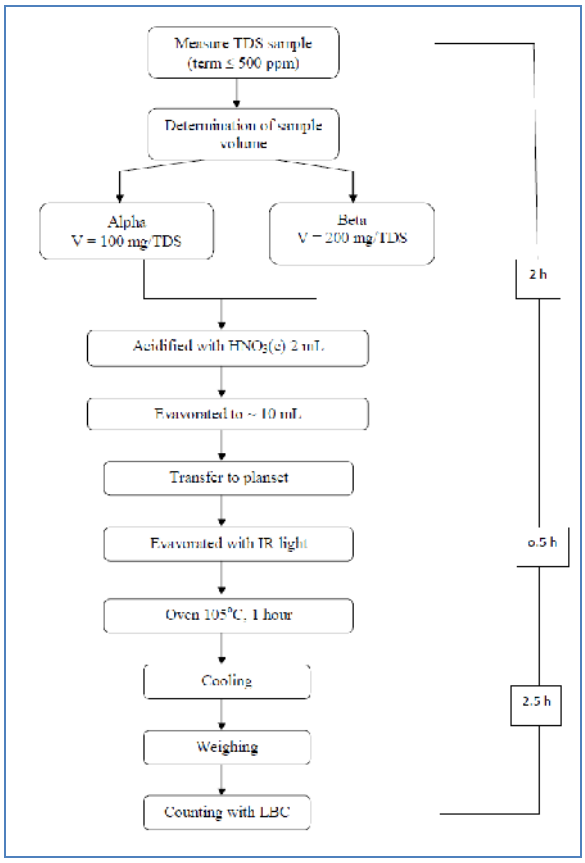

Figure 1. Flowchart of gross alpha and beta analysis based on EPA 900.0.

The mathematical formula used was as follow:

$$Efficiency\ of\ \alpha/\beta = \frac{(cs-cb)\ cps}{A\ dps} X\ 100\% \quad (1)$$

where :

cs : standard count rate (cps)
cb : background count rate (cps)
A : standard activity of α (^{241}Am)
or standar activity of β (^{90}Sr)

$$As = \frac{(cs-cb)\ cps}{E} \quad (2)$$

where :

As : Sample activity (Bq)
Cs : sample count rate (cps)
Cb : background count rate (cps)
E : efficiency

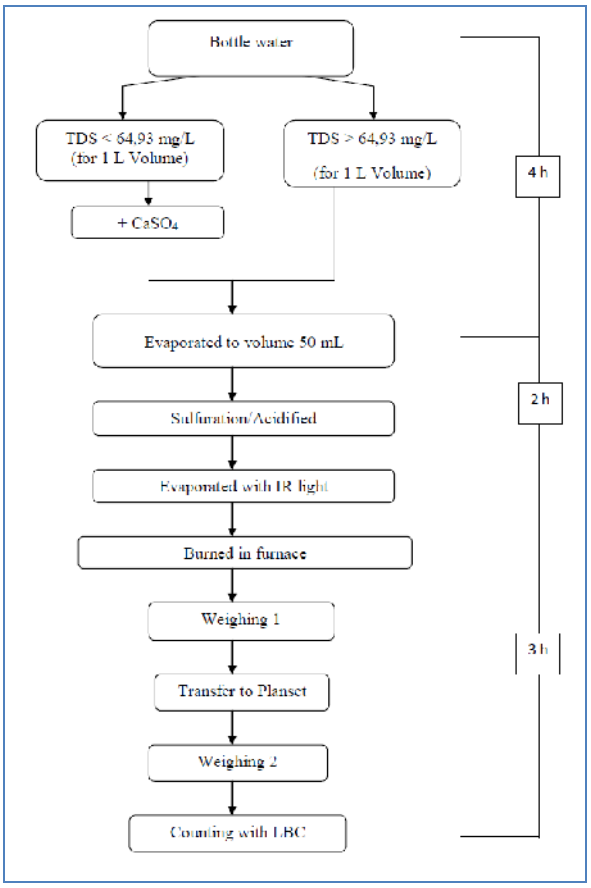

Figure 2. Flowchart of gross alpha and beta analysis based on SNI ISO 9696:2009 and SNI ISO 9697:2009.

The recovery of obtained activity for a sample was measured with the following formula :

$$R = \frac{As}{Astd} X\ 100\% \quad (3)$$

where:

R : *recovery* (%)
As : final activity obtained (Bq)
Astd : initial activity added (Bq)

Minimum detectable concentration (MDC) of LBC equipment was determinated by the following equation:

$$MDC = \frac{4{,}66\sqrt{\frac{cb}{tb}}}{E.V} X\ 100\% \quad (4)$$

where:

MDC : *minimum detectable concentration* (Bq/L)
cb : background count rate (cps)
tb : background count time (second)
E : efficiency of counting
V : volume of sample (L)

Results and Discussion

Calibration of measurement efficiency was performed to obtain a graph of the relationship between the thickness or weight of sediment with the radioactivity measured or counted so that the actual measurement of sample activity can be seen on the efficiency measurement as function of the thickness of

the sample used.

a. Determination of Calibration Curve Standars EPA 900.0 Method

Exponential regression calibration curve for the determination of alpha activity was obtained with the equation y = 0.211 $e^{-3.38x}$ with the correlation coefficient (R^2) of 0.914, and counting efficiency ranged from 14.96% to 21.26%. Exponential regression calibration curve for the determination of beta activity obtained with the equation y = 45.98 $e^{-0.89x}$, with the correlation coefficient (R^2) of 0.769, and counting efficiency ranged from 37.10% to 43.44%. For $^{90}Sr/^{90}Y$ + ^{241}Am mixed standard, exponential regression curve with the equation of y = 16.93 $e^{-0.94x}$, and the correlation coefficient (R^2) of 0.694 for alpha and the equation y = 44.13 $e^{-0.53x}$, the correlation coefficient (R^2) of 0.815 for beta. The efficiency of alpha counting (^{241}Am) was ranged 15.58% to 16.57% and for beta counting ($^{90}Sr/^{90}Y$) was ranged from 41.78% up to 43.44%.

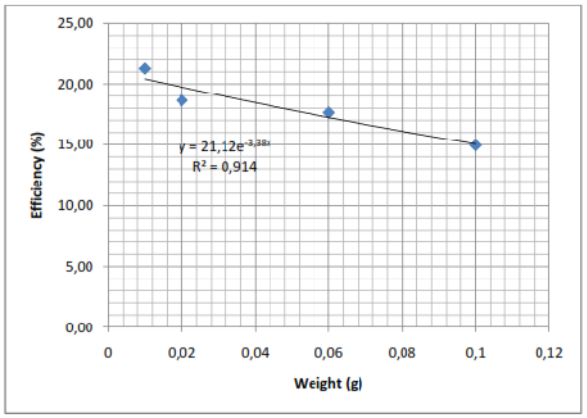

Figure 3. Graph of sample weight and ^{241}Am standar efficiency relationship.

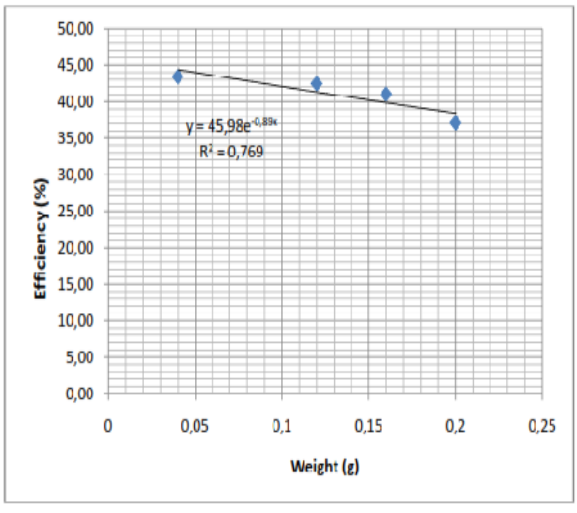

Figure 4. Graph of sample weight and $^{90}Sr/^{90}Y$ standard efficiency relationship.

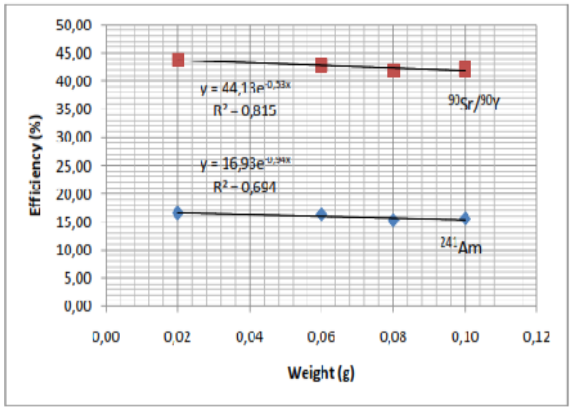

Figure 5. Graph of sample weight and ^{241}Am+ $^{90}Sr/^{90}Y$ mixed standard efficiency relationship.

b. Determination of the method of standard calibration curve SNI ISO 9696:2009 and SNI ISO 9697:2009

Regression exponential calibration curve to determine alpha activity obtained with the equation y = 5.675 $e^{-1.73x}$, the correlation coefficient (R^2) of 0.937, and alpha counting efficiency ranged from 0.74% to 5.35%. Exponential regression calibration curve for the determination of beta activity obtained with the equation y = 39.06 $e^{-0.43x}$, the correlation coefficient (R_2) of 0.994, and beta counting efficiencies ranged from 23.05% to 37.01%. For $^{90}Sr/^{90}Y$ + ^{241}Am mixed standard, the obtained exponential regression curve with the equation y = 6.74 $e^{-1.90x}$, the correlation coefficient (R^2) of 0.966 for alpha and the equation y = 41.89 $e^{-0.67x}$, the correlation coefficient (R^2) was 0.917 for beta. The efficiency of counting of alpha (^{241}Am) is obtained in the amount of 0.73% to 5.82% and for beta ($^{90}Sr/^{90}Y$) obtained efficiency was from 17.02% to 35.17%.

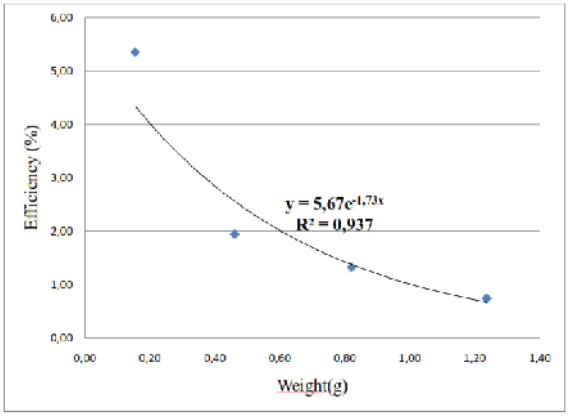

Figure 6. Graph of sample weight and ^{241}Am standard efficiency relationship.

Efficiency calibration results as a function of weight (residual water) as shown in the graph above shows the counting efficiency decreased with the

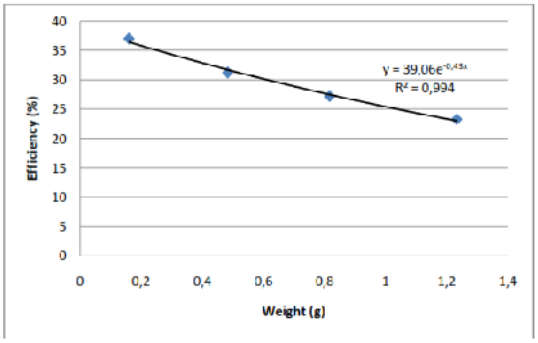

Figure 7. Graph of sample weight and $^{90}Sr/^{90}Y$ standard effieciency relationship.

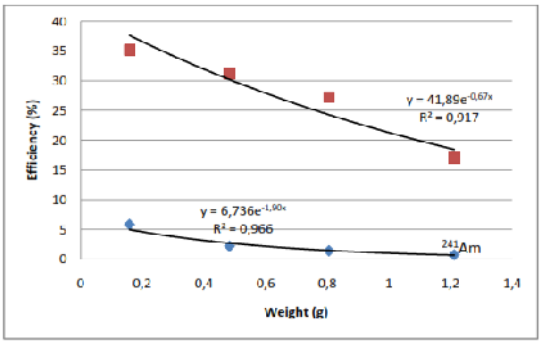

Figure 8. Graph of sample weight and $^{241}Am+^{90}Sr/^{90}Y$ standard efficiency relationship.

increasing weight of sediment in planset caused by self absorption factor of the sediment (matrix). Self absorption factor relies heavily on the thickness of sediments; the thinner the sediment, the smaller the self absorbtion. Self absorption factor is due to the small penetrating power, especially for alpha radiation. The smaller the power breakdown, the greater the absorption factor itself.

c. Comparison method of EPA 900.0 with Metode SNI ISO 9696:2009 and SNI ISO 9697:2009.

Comparison of EPA method 900.0 with method SNI ISO 9696:2009 and SNI ISO 9697:2009 was done by considering some aspects, especially the recovery of the determination of alpha and beta radioactivities in water samples and the total preparation time required.

Determination of gross alpha and beta activities was performed 5 times for each standard solution. Determination results and recovery activities of both methods can be seen in Table 1 and 2.

In Table 1 it is shown that in the EPA method 900.0, the average recovery on counting of ^{241}Am alpha using the standard calibration was 59.31%. Average recovery for beta using $^{90}Sr/^{90}Y$ standard as standard calibration was 92.90%. Median recovery of alpha counting using a mixed standard of $^{241}Am+^{90}Sr/^{90}Y$ as standard calibration was 41.87%. The average recovery in the counting of beta using the mixed standard of ^{241}Am + $^{90}Sr/^{90}Y$ as calibration standards was 92.26%.

Table 2 shows that in the method SNI ISO 9696:2009 and SNI ISO 9697:2009 the obtained average recovery on counting of ^{241}Am alpha using the standard calibration standards was 10.22%. The average recovery for beta using $^{90}Sr/^{90}Y$ standard as calibration standards was 72.46%. The average recovery of alpha counting using a mixed standard of $^{241}Am+^{90}Sr/^{90}Y$ as calibration standards was 10.94%. Average recovery on the counting of beta using the mixed standard of $^{241}Am+^{90}Sr/^{90}Y$ as calibration standards was 81.78%.

In Table 1 and 2 shows that the general recovery of alpha activity is smaller than the beta, this was due to the penetrating power of beta particles were higher than the alpha particles that can penetrate in deeper sediment (matrix) samples and detected by the detector. This is what the cause the maximum sample weight limits for EPA methods 900.0, ie for a total of 0.100 g of alpha and a total of 0.200 g for beta. As for the method of SNI ISO 9696:2009 and SNI ISO 9697:2009 sample weight should be between 0.1734 g and 1.0000 g.

Table 1. Activity and recovery of EPA 900.0 method.

No	Alpha counting (Source ^{241}Am)		Beta counting (source $^{90}Sr/^{90}Y$)		Alpha counting (source $^{241}Am+^{90}Sr/^{90}Y$)		Beta counting (source $^{241}Am+^{90}Sr/^{90}Y$)	
	Activity (Bq)	*Recovery* (%)	Activity (Bq)	*Recovery* (%)	Activity (Bq)	*Recovery* (%)	Activity (Bq)	*Recovery* (%)
1	5.86	41.38	8.03	80.30	5.59	39.49	9.60	96.04
2	7.33	51.76	9.95	99.54	5.39	38.09	9.07	90.74
3	9.68	68.33	9.95	95.22	5.44	38.42	8.06	80.64
4	9.05	63.93	9.01	90.11	6.89	48.68	9.72	97.15
5	10.00	71.13	9.93	99.34	6.33	44.68	9.67	96.73
Average	9.37	59.31	9.37	92.90	5.93	41.87	9.22	92.26

Note : MDC alpha 0.02 Bq/L dan MDC beta 0.01 Bq/L

Table 2. Activity and recovery of SNI ISO 9696:2009 and SNI ISO 9697:2009 methods.

No	Alpha counting (Source ^{241}Am)		Beta counting (source ^{90}Sr/^{90}Y)		Alpha counting (source ^{241}Am+^{90}Sr/^{90}Y)		Beta counting (source ^{241}Am+^{90}Sr/^{90}Y)	
	Activity (Bq)	*Recovery* (%)	Activity (Bq)	*Recovery* (%)	Activity 9Bq)	*Recovery* (%)	Activity (Bq)	*Recovery* (%)
1	1.35	9.53	7.38	73.81	1.36	9.63	7.68	76.78
2	1.22	8.59	6.64	66.36	1.58	11.17	8.21	82.10
3	1.63	11.49	7.07	70.74	1.65	11.65	8.17	81.67
4	1.31	9.28	7.71	77.07	1.41	9.98	8.93	89.30
5	1.73	12.22	7.43	74.30	1.74	12.26	7.90	79.04
Average	1.45	10.22	7.25	72.46	1.55	10.94	8.18	81.78

Note : MDC alfa 0.02 Bq/L dan MDC beta 0.01 Bq/L

In the comparison of recovery of alpha and beta activity between EPA 900.0 method with SNI ISO 9696:2009 and SNI ISO 9697:2009 method showed that recovery using EPA 900.0 method was greater than the SNI ISO 9696:2009 and SNI ISO 9697:2009. It also can be seen from the time of analysis process that it was shorter so that the loss of activity during the process can be reduced.

The time required for EPA 900.0 method in determining the activity of alpha and beta was shorter with total of about 6 hours whereas the time required by SNI ISO 9696:2009 and SNI ISO 9697:2009 was approximately 9 hours. In addition in the case of the use of materials and equipment, EPA method is simpler than the method of SNI ISO 9696:2009 and SNI ISO 9697:2009. In Table 3, we can see the summary of comparison of the two methods.

Table 3. Comparison of EPA 900.0 with SNI ISO 9696:2009 and SNI ISO 9697:2009 methods.

No	Parameter	EPA 900.0	SNI ISO 9696:2009 dan SNI ISO 9697:2009
1	Recovery : - Apha * - Beta ** - Alpha*** - Beta***	 59.31 % 92.90 % 41.87 % 92.26 %	 10.22 % 72.46 % 10.94 % 81.78 %
2	Time	6 h	9 h
3	Material and apparatus	Less	More

Note :
* Calibration use ^{241}Am source
** Calibration use ^{90}Sr/^{90}Y source
*** Calibration use mixed ^{241}Am+^{90}Sr/^{90}Y source

Conclusion

In the EPA 900.0 method, the average recoveries of alpha counting (^{241}Am standard) was 59.31%, of beta counting (^{90}Sr/^{90}Y standard) was 92.90%, and of alpha counting (^{241}Am + ^{90}Sr/^{90}Y standard) was 41.87%, and of beta counting (^{241}Am + ^{90}Sr/^{90}Y standard) was 92.26%. In SNI ISO method 9696:2009 and ISO 9697:2009, the average recoveries of alpha counting (^{241}Am standard) was 10.22%, of beta counting (^{90}Sr/^{90}Y standard) was 72.46%, of alpha counting (^{241}Am + ^{90}Sr/^{90}Y standard) was 10.94 %, and of beta counting (^{241}Am + ^{90}Sr/^{90}Y standard) was 81.78%. Based on the time analysis, EPA method needed $\pm$ 6 hours while the method of SNI ISO 9696:2009 and SNI ISO 9697:2009 needed $\pm$ 9 hours, and based on materials and equipment, EPA 900.0 method was simpler, so that the overall EPA 900.0 method was more effective and efficient.

Acknowledgements

We would like to thank the Center for Technology of Radiation Safety and Metrology – BATAN, especially Safety and Health Division for all the support given.

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DISCUSSION

1. Gatot Wurdiyanto, PTKMR BATAN:

   Question: Did you measure or determine the effect of alpha counting when counting the beta?
   Answer : In this research we did not do the correction "crosstalk", but in the next time we will do it.

2. Nazaroh, PTKMR BATAN:

   Question: Which is better, EPA 900.0 method or ISO SNI method ?
   Answer : By considering the recovery obtained it can be concluded that EPA 900 method is better than ISO SNI where EPA method is shorter so that there is small loss of activity.

Standardization of Radioactive Solution of ^{133}Ba Using Gamma Spectrometry Methods

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Abstract. **A standard radioactive solution of ^{133}Ba has been created by using gamma-ray spectrometry method at the Center for Technology of Radiation Safety and Metrology (PTKMR) - BATAN. This creation need to be done in order to get master solution of radioactive standards of ^{133}Ba that will be used as a standard source to calibrate the nuclear instruments. ^{133}Ba radioisotope was obtained from the manufacturers of POLATOM - POLAND as the raw material and processed to be the standard radioactive sources. The source of ^{133}Ba was diluted by a certain dilution factor and then 27 pieces of the sample were made as point sources in a thin layer of Mylar. Preparation method used was gravimetry because it was more accurate than other methods. Radioactivity measurements were conducted using gamma-ray spectrometry. Measurements of source impurities were carried out simultaneously and individually with the measurement of radioactivity. Standard source used was multi-gamma source of ^{152}Eu that had traceability to the International System of Units. The measurement was corrected by dead time, decay, background, half life, impurity, and the effect of the sum spectrum (sum-peak). Determination of homogeneity factor of ^{133}Ba radioactive solution was determined by the amount of radioactivity per weight unit. The value of the homogeneity factor was quite good with the level of acceptability below 3% and the specific activity of ^{133}Ba solution was (144 ± 3) Bq/mg on June 1, 2011 at 12:00 WIB. By this research it is expected that the Radiation Metrology Laboratory, PTKMR-BATAN can make a solution of ^{133}Ba standard sources that have a sufficient level of homogeneity and have traceability to the International System of Units, so that the utilization of nuclear technology in all fields can be accomplished with good safety and security for workers, communities and the environment.**

Keywords: Homogeneity, Barium 133, a standard source, and gamma spectrometry.

Introduction

Utilization of nuclear technology in all fields has grown rapidly in the world and the utilization of a standard needs technical requirement so that the product has an adequate level of quality. Some laboratories use a standard to calibrate equipment owned, as well as the as a e quality assurance of the product. One of the requirements that are used to obtain a qualified reference standard material will be required an adequate degree of homogeneity.

Based on the duties and functions as a national reference laboratory in the field of radioactivity measurements, Center for Technology of Radiation Safety and Metrology (PTKMR) should be able to provide a standard source of different types so that the results of measurement and testing of samples had values that are accurate, precise and traceable to the International System of Units.

This research has developed a fabrication method of a reference standard solution of barium-133 (or ^{133}Ba). Raw material of ^{133}Ba obtained from radioisotopes made by POLATOM – POLAND will be created a ^{133}Ba master solution. Raw material solution of ^{133}Ba was prepared and made some point sources and then standardized using gamma-ray spectrometry method. This method is chosen because it is very flexible for measuring radioactivity and emits gamma photons can be analyzed qualitatively and quantitatively. In addition this method can be used to analyze the impurities of these standard sources, so that when the impurity is detected then the correction can be performed simultaneously with measurements of radioactivity source. Measurement of activity for ^{133}Ba has been done by analyzing the energy spectrum at 356.46 keV.

Some researchers has standardized ^{133}Ba sources using 4πβ-γ coincidence method (Miyahara and Mori, 1990; Kawada and Hino, 1972). By this method, standardization of ^{133}Ba can be directly performed because ^{133}Ba decays by electron capture model and emitting gamma simultaneously (ICRP, 1983; Table de Radionucleides, 1982; Kri et al., 2004; Ineel et al., 2010). A ^{133}Ba radionuclide has very long half life (3,846 days) (Ineel and Kri, 2007), decays by electron capture model while emitting gamma photons at energies of 53-384 keV with the greatest intensity about 62% at 356 keV energy, to be a stable element of ^{133}Cs (Ineel, 2010). With these characteristics, ^{133}Ba is an excellent standard source for gamma photons that emitting at energies below 500 keV.

The purpose of this study was to obtain a reference standard material of radioactive solution of ^{133}Ba which has a sufficient level of homogeneity and its activity value has a traceability to the International

System of Units, so the need for a national radioactive standard sources can be obtained independently. With the standard sources are expected that the use of nuclear technology in all fields can be accomplished with a secure and safe for workers, community members and the environment.

Materials and Methods

Preparation

Before sample preparation was made, the container, tools and support resources in the form of ampoules or thin film layer of Mylar were cleaned and sterilized from any impurity elements. Preparation was carried out by gravimetric method using a calibrated semi-micro balance device because it is more accurate than other methods. Twenty seven pieces of point sources were prepared. All of the ^{133}Ba sources were dripped on mylar buffer and then dried using infra-red lamp. The dried sources were covered with a Mylar layer and coded and ready to be measured.

Measurement of activity

The activities of the point sources were measured by a Gamma Spectrometer (Figure 1). The detector was a HPGe model GC1018 (Canberra, USA), which had a relative efficiency of 10.3% with an energy resolution of 1.69 keV FWHM at 1332.5 keV (Knoll, 1989; Debertin and Helmer, 1988). The detector was equipped with a model 2002CSL pre-amplifier, and a Canberra model 2020 amplifier and operates at a bias voltage of +4500 V. Signals from the detector were processed by the Canberra gamma spectrum analysis system using GENIE 2000 software (Canberra Industries, USA). The source-to-detector distance was 25 cm. As described above, the gamma-ray spectrometry system was first calibrated using standard sources of ^{152}Eu that had traceability to the System of International. Three sets of measurements were made with a counting time of 7,200 seconds in each set. Measurement of standard sources made during the 10,000 seconds in order to achieve high accuracy value.

Measurement of impurity

Impurity was measured using a gamma-ray spectrometer similar to that used when measuring the activity. This spectrometer could perform qualitative and quantitative analysis. Measurement was carried out for several times with a longer time than measurement of activity to know the possibility of a very small impurity in the sample being measured. Standard source used in the measurement of this impurity was ^{152}Eu. Impurity content of a standard material needed to be known and must be corrected in order to obtain accurate values and rigorous standards. Measurement of impurity should be done separately so that the sum-peak and pile-up effects, which detected in a minimum level. These factors were a major nuisance when performing qualitative analysis.

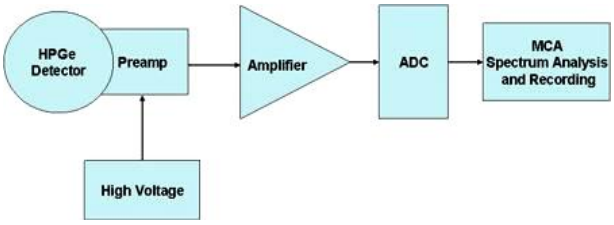

Figure 1. Block Diagram of gamma-ray spectrometer device with HPGe detector.

Correction

A number of corrections that affected the ideal conditions of measurement was performed included background, dead time, sample impurities, decay factors, and others (Susetyo, 1988). Correction for background was done by measuring the background counts for 54,000 seconds. This was intended to get an accurate measurement value. The correction for dead time was made directly by setting a counting time on a live-time position.

Results and Discussion

The weighing results of the ^{133}Ba source were presented in Table 1. The number of sources that dripped into the Mylar buffer was adjusted to a diameter of less than 0.5 cm and the estimated activity did not exceed the maximum limits of the equipment.

The result of the efficiency calibration using the standard source of ^{152}Eu is shown in Fig. 2. The energies used to produce the efficiency calibration curve were 244.7, 344.3, 778.9, 964.1, 1112. and 1408.0 keV, resulting in a curve described by the equation of Efficiency at energy $E = 0.1485\ e^{-0.971}$ with a correlation coefficient of 0.9996. The uncertainty of the interpolated efficiency at any point was 1.5 % at k = 1. This value was determined from the residuals between the true efficiencies and the efficiencies obtained from the efficiency curve. From this curve the efficiency value obtained was 0.0004946 for energy of 356.0129 keV.

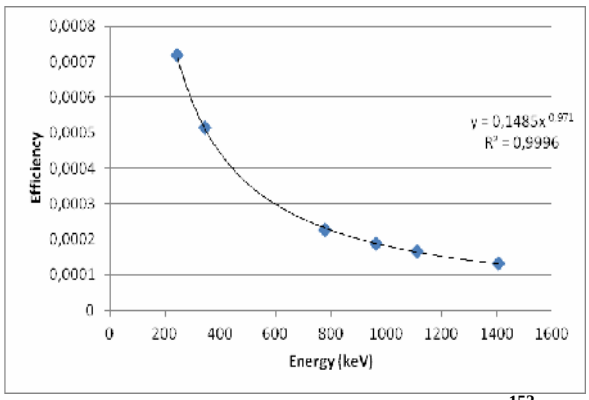

Figure 2. Efficiency calibration curve using ^{152}Eu standard source.

Table 1. Preparation data of ^{133}Ba.

No.	Code No.	Weight (mg)	No.	Code No.	Weight (mg)
1	13302/11	8.48	15	13316/11	13.49
2	13303/11	13.72	16	13317/11	18.30
3	13304/11	13.08	17	13318/11	19.94
4	13305/11	14.51	18	13319/11	18.79
5	13306/11	13.05	19	13320/11	18.21
6	13307/11	12.63	20	13321/11	17.02
7	13308/11	15.24	21	13322/11	26.92
8	13309/11	12.67	22	13323/11	20.36
9	13310/11	10.75	23	13324/11	21.26
10	13311/11	11.67	24	13325/11	23.25
11	13312/11	10.92	25	13326/11	13.76
12	13313/11	11.35	26	13327/11	14.69
13	13314/11	14.31	27	13328/11	11.96
14	13315/11	14.93			

The spectrum of ^{133}Ba shown in Figure 3 was obtained by using the gamma spectrometer with HPGe semiconductor detectors. In this spectrum a significant impurity was not found, so that the correction to the impurity is not necessary.

The result of the determination of the activity of ^{133}Ba source of each source is shown in Table 2. Because the source was derived from the same solution the specific activity of the solution from that source can be calculated. Value of specific activity of a solution of ^{133}Ba was done by dividing the weight of each source that dripped on each sample. The results of the determination of the activity for each samples were shown in Table 2, with an average value (144.23 ± 2.75) Bq/mg. These results were quite good because it had a standard deviation below 3%. Specific activity values obtained through measurements of each sample was made from the original liquid, which had a value of the uncertainty of less than 3%, it could be said that the original liquid was fairly homogeneous with the limit of acceptability of 3%.

To determine the value of standard uncertainty of the activity was carried out by determining the various components that affected the calculation of the activity. Table 3 showed some components to be reckoned in determining the uncertainty of a stretch for the measurement of radioactivity using gamma-rays spectrometry method. From these components it can be determined that the value of the expanded uncertainty of ^{133}Ba source solution is 3.8% with 95% of confidence level, and coverage factor, k was = 2. So the value of the solution activity of ^{133}Ba source was 144.23 Bq / mg, on June 1, 2011 at 12:00 pm.

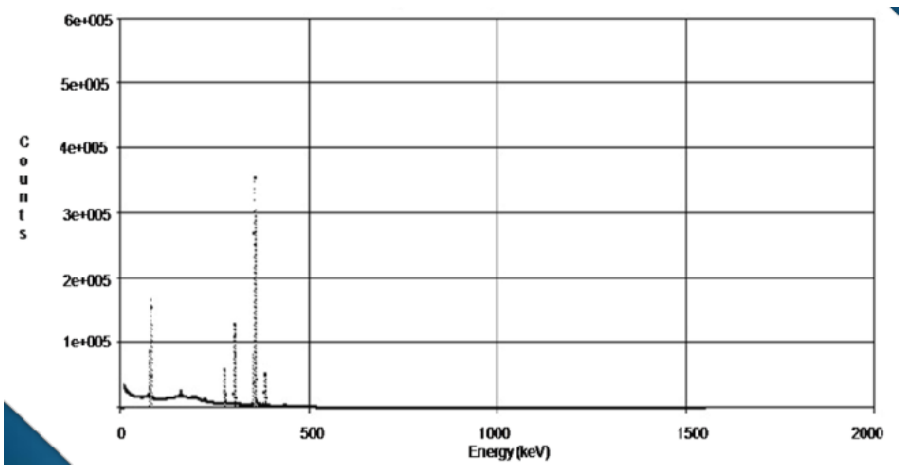

Figure 3. Spectrum of ^{133}Ba using Gamma-ray Spectrometer System with Semiconductor HPGe detector.

Table 2. Results of ^{133}Ba activity measurement (reference time : June 1st 2011, at 12.00 WIB).

No.	Sample code	Weight (mg)	Activity (Bq)	Specific Activity (Bq/mg)
1	13302/11	8.48	1266	149.34
2	13303/11	13.72	2025	147.54
3	13304/11	13.08	1942	148.41
4	13305/11	14.51	2094	144.33
5	13306/11	13.05	1869	143.19
6	13307/11	12.63	1873	148.24
7	13308/11	15.24	2191	143.80
8	13309/11	12.67	1837	144.98
9	13310/11	10.75	1536	142.90
10	13311/11	11.67	1732	148.36
11	13312/11	10.92	1592	145.75
12	13313/11	11.35	1645	144.90
13	13314/11	14.31	2126	148.59
14	13315/11	14.93	2148	143.88
15	13316/11	13.49	1986	147.16
16	13317/11	18.30	2556	139.67
17	13318/11	19.94	2856	143.22
18	13319/11	18.79	2606	138.71
19	13320/11	18.21	2635	144.69
20	13321/11	17.02	2522	148.20
21	13322/11	26.92	3828	142.21
22	13323/11	20.36	2920	143.40
23	13324/11	21.26	3040	142.97
24	13325/11	23.25	3345	143.86
25	13326/11	13.76	1937	140.35
26	13327/11	14.69	2086	142.06
27	13328/11	11.96	1711	143.03
	Mean value			144.23 ± 2.75

Table 3. Uncertainty components for ^{133}Ba activity measurement using the Gamma-ray spectrometer.

Source of uncertainty	Standard uncertainty components (%)	
	Type A	Type B
Standard Source		0.5
Half- life of standard		0.12
Efficiency		1.5
Intensity of sample		0.1962
Half life of sample		0.0164
Area of sample	0.67	
Dead time		0.05
Quadratic sum	0.67	1.76
Combined standard uncertainty	1.9	
Expanded Uncertainty (k = 2)	3.8	

Conclusions

Master solution of ^{133}Ba obtained was fairly homogeneous with acceptance limits below 3%. The results of activity measurements of each sample were quite good with a expanded uncertainty of 3.8% with the covered factor, k = 2. Value of specific activity of the master solution ^{133}Ba was 144.23 Bq/mg, on June 1, 2011 at 12.00 pm. Standardization in liquid form of ^{133}Ba source can be performed at the National Metrology Radiation Laboratory, and a solution of ^{133}Ba radioactive sources can be used as a reference standard material.

Acknowledgements

The authors would like to thank the Head of Center for Technology of Radiation Safety and Metrology that gives us the opportunity to conduct this study so that we get a chance to standardize ^{133}Ba source.

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Analysis of the TENORM Radiation Safety on Several Offshore of Oil and Gas Industries in Java Island

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Abstract. **Analysis of TENORM radiation safety on several offshore of three oil and gas industries (A, B and C) in Java Island has been carried out. Analysis included radiation exposure rate, surface contamination, the kind and level of concentration of radionuclides in TENORM. Radiation exposure was measured by using Ludlum 19 surveymeter and surface contamination was measured by using the Mini Monitor Series 900 at ±5 mm distance from the surface of TENORM. While the kind and level of radionuclides in TENORM concentration were determined by collecting the TENORM samples that were measured by using the ORTEC's GEM-60HPGe Gamma Spectrometer. The result of radiation exposure rate in oil and gas industries A, B and C was ranged from not detected to 38.0 ± 2.8 µR/h, 42.5 ± 4.9 µR/h and135.0 ± 7.1 µR/h, respectively. The surface contamination in oil and gas industries A, B and C was ranged from not detected to1.88 ± 0.30 Bq/cm^2, 1.3 ± 0.09 Bq/cm^2 and 0.74 ± 0.05 Bq/cm^2, respectively. While the kind and level of radionuclide concentrations were ranged from 0.14 ± 0.02 Bq/kg to 18,450 ± 1,766 Bq/kg for ^{232}Th radionuclide, 0.12 ± 0.01 Bq/kg to 18,291 ± 1,738 Bq/kg for ^{228}Th radionuclide, 0.06 ± 0.01 Bq/kg to 9,507 ± 894 Bq/kg for ^{226}Ra radionuclide and 0.09 ±0.01 Bq/kg to 1,545 ± 218 Bq/kg for ^{40}K radionuclide. According to the terms of the Nuclear Energy Regulatory Agency (BAPETEN) that the value of radiation exposure rate, surface contamination and levels of radionuclide concentration at several measurements point in the TENORM of three oil and gas industries examined have exceeded the permitted limit values. Therefore, they need to be the remedial action and manage the working duration for workers so that the public-dose limit values are not exceeded.**

Keywords: radiation safety, TENORM radiation, oil and gas industry

Introduction

The mining industry is basically an attempt by the utilization of natural resources contained within the bowels of the earth whose activities include excavation (exploitation) and mineral processing to obtain the desired product. In the treatment process, the natural radioactive elements contained in the earth's crust will be mobilized and eventually will be concentrated during the treatment process. The contents of radioactive material as a result of the follow-up process is called TENORM (Technologically Enhanced Naturally occurring Radioactive Materials) that is due to the natural radioactive in human activity or process technology that increased potential exposure when compared with the early condition (Perka BAPETEN No. 9, 2009).

TENORM can contribute to external and internal radiation exposures of workers to the community even in the vicinity of the industry. Workers will receive external radiation exposure that is near industrial facilities that have been contaminated with TENORM. Whereas internal radiation worker will be accepted through the respiratory tract, oral or skin when air, gas or TENORM contaminated material enters into the body. Therefore the safety analysis of the TENORM radiation needs a serious attention.

According to Indonesian Government Regulation No. 27 Year 2002 on Radioactive Waste Management that every person or entity that performs nonnuclear mining minerals that can produce radioactive waste as a byproduct of mining required to perform radiation safety analysis (PP-RI No. 27, 2002). According to Government Regulation No.33 of 2007 on Ionizing Radiation Safety and Security of Radioactive Sources that all mineral and oil and gas industries required to monitor and manage the the entire area of TENORM operation (PP-RI No. 33, 2007). According to Regulation of the Nuclear Energy Regulatory Agency (BAPETEN) No. 9 Year 2009 on the Intervention of the exposure that comes from TENORM that producers should do TENORM radiation safety analysis in each locations owned or were in its control includes the type and the activities carried out, the amount or quantity of TENORM, the type and level of radionuclide concentrations and exposure to radiation and/or contamination of the highest in the surface TENORM (Perka BAPETEN No. 9, 2009).

In an effort to meet regulatory requirements and to protect workers, the environment and society, we analyzed the TENORM radiation safety in several offshore oil and gas industry in Java island included exposure to radiation and/or contamination of the highest in the surface TENORM, and the type and

level of radionuclide concentrations, so that it can be taken to reduce or eliminate the hazards of TENORM radiation for workers, the environment and the general public.

Natural radioactivity in the Earth layer

In the earth there is a natural radioactive substance which is a type of primordial radioactive nature that has been formed since the formation of earth. Radioactive substances included in this type are ^{238}U along with its daugthers known as ^{238}U series, ^{232}Th radioactive substance along with its daugther known as ^{232}Th series and ^{40}K radioactive substances. ^{238}U and ^{232}Th has a very long half life (billions of years) and the decay process will produce a variety of daugthers shed with half life of the order of seconds to thousands of years as in Table 1.

Natural radioactive properties in several types of soil and rock that contained deep within the earth varied as shown in Table 2. ^{226}Ra concentrations ranged from 5-1800 Bq/kg and ranged from 4-540 Bq/kg for ^{228}Th. ^{40}K concentrations are generally higher for all types of soil and rock than the content of ^{226}Ra and ^{228}Th. ^{226}Ra concentration in shale rock and gneiss rock is higher than the other. While the ^{228}Th highest concentration in granitic rocks, limestone and gneis (Wiharto and Syarbaini, 2003; UNSCEAR, 1999).

Table 1. ^{238}U and ^{232}Th series with their daughters (Wiharto and Syarbaini, 2003).

Nuclides	Half Life	Decay Mode
^{238}U	4.5 x 10^{9} years	Alfa
^{234}Th	24,10 days	Beta
^{234m}Pa	1.2 minutes	Beta
^{234}U	2.5 x 10^{5} years	Alfa
^{230}Th	7.7 x 10^{4} years	Alfa
^{226}Ra	1.6 x 10^{3} years	Alfa
^{222}Rn (gas)	3.8 days	Alfa
^{218}Po	3.05 menutes	Alfa
^{214}Pb	26.8 menutes	Beta
^{214}Bi	19.9 menutes	Beta
^{214}Po	1.6 x 10^{-4} seconds	Alfa
^{210}Pb	22.3 years	Beta
^{210}Bi	5 days	Beta
^{210}Po	138 days	Alfa
^{206}Pb	Stable	

Nuclides	Half Life	Decay Mode
^{232}Th	1.4 x 10^{10}years	Alfa
^{228}Ra	5.8 years	Beta
^{228}Ac	6.13 hours	Beta
^{228}Th	1.9 years	Alfa
^{224}Ra	3.64 days	Alfa
^{220}Rn (gas)	55 seconds	Alfa
^{216}Po	0.15 secons	Alfa
^{212}Pb	10.6 hours	Beta
^{212}Bi	60.6 second	Alfa, beta
^{208}Tl	3.1 menutes	Beta
^{212}Po	3 x 10^{-7} seconds	Alfa
^{208}Pb	Stable	

TENORM Procces in the Oil and Gas Industry

In the oil and gas industry there are two main types of by-product that formed from the production of scale and sludge. Scale is sulfate and carbonate salts deposited on the surface of the means of production that are $BaSO_4$, $CaSO_4$ and $CaCO_3$. While the sludge is a mixture of organic compounds and solid minerals in the water used to separate liquid hydrocarbons from oil and gas production equipment. Scales that formed from process of the compounds of uranium and thorium are generally not soluble in water, so that they left on the underground cistern (underground reservoir) at the time of the oil, gas and water flowing to the surface of the earth. Some compounds of radium and its decay daugthers are

Table 2. Concentration of natural radioactive substances in various types of soil and rock (Bq/kg) (Wiharto and Syarbaini, 2003; UNSCEAR, 1993).

Types of soil and rock	^{228}Ra (^{232}Th)	^{226}Ra (^{238}U)	^{40}K
Sand and silt	4 - 30	5 - 25	600 - 1200
Clay	25 - 80	20 - 120	600 – 1300
Moraine	20 - 80	20 - 80	900 – 1300
Soil – Aluminum shale	20 - 80	100 - 1000	60 – 1000
Normal Granite	0.4 - 103	1 - 370	600 - 1800
Th and U rich granite	40 - 350	100 - 500	1200 – 1800
Shale	0.2 - 60	0.4 - 120	600 – 1800
Limestone	0.1 - 540	0.4 - 340	30 – 150
Sandstone	4 - 40	5 - 60	300 – 1500
Diorite	4 - 40	1 - 20	300 – 1000
Gneis	0.4 - 420	1 - 1800	600 - 1800

soluble in water when the water was taken out. Because of the influence of temperature, pressure and pH, the elements of Ba and Ca will be settled in $BaSO_4$, $CaSO_4$ and $CaCO_3$ forms. Because of the chemical nature of Ra with Ba and Ca (both class IIA) then Ra will also be settled. Ra concentration in the scale varies depending on the geological structure of the place of origin.

Sludge formation process is similar to the formation of scale. Due to changes in temperature and pressure, dissolved solids will settle in the production system. The precipitates are formed in the form of oily material and flexure. Sludge often contains silica compounds but sometimes also contains barium. Concentration of radionuclides in the sludge varies with the highest concentration in the separator and the surrounding areas.

Scale and sludge are commonly found in the means of production such as tubular, wellhead, valves, pump and separator. Scale in the oil industry generally is in the form of barium sulphate. Precipitate of radium with barium and then replace some of barium atoms in the crystal structure of barium sulfate. Therefore ^{228}Th decays into ^{228}Ra and ^{226}Ra decays into ^{210}Pb radon gas through (^{222}Rn), the scale was found also in the ^{228}Th and ^{210}Pb. While many in the natural gas found in radon gas which is the whole of ^{226}Ra. At this stage of oil separation with gas and water, radon gas will follow the natural gas. However, radon gas decay life has short, so it is found accumulated in the means of production of natural gas the decay daughter is ^{210}Po and ^{210}Pb in the form of a thin layer of such films.

If sea water is used as the water injection water is injected into the reservoir to increase oil recovery, the sea water mixed with formation water can add content to the deposition of sulfate scale will also increase.

Analysis of theTENORM Radiation Safety

Analysis of theTENORM radiation safety included: exposure to radiation and/or contamination of the highest in the surface TENORM, the type and concentration of radionuclides, and the type of activities carried out and the amount or quantity TENORM (Perka BAPETEN No. 9, 2009). To perform the analysis it is required radiation safety measuring instrument sand nuclear spectroscopy. Radiation exposure is measured directly using the radiation gauge surveimeter and calculated by the equation:

$$D = \frac{D_R C_F}{A_F}(\mu R/h) \qquad (1)$$

where :

D: gamma exposure rate (μR/h)
D_R: Reading measure instruments of surveymeter (μR/h)
C_F: calibration factor
A_F: absorption correction factor, depending on the type of TENORM

TENORM surface contamination was measured directly by using a contamination monitor instrument and is calculated by the following equation:

$$SC = \frac{(N_t - N_b)C_F}{A_F}(Bq/cm^2) \qquad (2)$$

where :

N_t: total initial count rate (cps)
N_b: background count rate (cps)
C_F: calibration factor (Bq/cm^2/cps)
A_F: absorption correction factor (<1), depend on the material and surface quality.

The type and concentration of radionuclides in TENORM levels were measured using the nuclear spectroscopy system with a gamma spectrometer detector of high purity (HPGe: High Purity Detector). Before being used in the measurement, a device must be calibrated gamma spectrometer to be used for analysis. There are two kinds of calibrations performed the energy calibration and efficiency calibration. Energy calibration is required for the purposes of qualitative analysis, while the efficiency calibration for quantitative analysis purposes (Susetyo, 1988).

Calibration is done by comparing the efficiency of the detected gamma radiation with a standard amount of a radioactive source activity. Counting efficiency is determined by the following equation (Wiyono and Bunawas, 2007, Perka BATAN, 1998) :

$$\varepsilon = \frac{N}{At.P\gamma}\ x\ 100 \qquad (3)$$

where N : standard source count rate(cps)
At : the current standard source activity measurement(Bq).
Pγ : gamma energy abundance(%)

To calculate the standard source activity at the time of measurement we used equation above: [7, 8]

$$A_t = A_o.e^{-\frac{0,693}{T_{1/2}}.t} \qquad (4)$$

where A_0 : initialactivity (Bq)
$T_{1/2}$: half-life radionuclides (years)
t : the time between the time of the determination of the activity until the time of measurement (year).

By calculating the efficiency of the enumeration of each energy source that can be made standard efficiency calibration curve that then used to

calculate the activity of radionuclide. Activity of radionuclide concentrations calculated by the equation (Wiyono and Bunawas, 2007; Perka BATAN, 1998):

$$C = \frac{(N_t - N_b)}{\varepsilon . P\gamma . Fk . W} \quad (5)$$

where C : radionuclide activity concentration (Bq/kg)

N_t : count rateof sample (cps)
N_b : background count rate(cps)
ε : counting efficiency (efficiency is determined from calibration curve)
Pγ : gamma energy abundance (%)
Fk : self-absorption correction factor.
W : weight sample (kg).

Self-absorption correction factor calculated when there is a difference between the density of samples with source density standar (Wiyono and Bunawas, 2007; Perka BATAN, 1998).

Measurement uncertainty for the measurement of radionuclide activity concentration (μC) with a confidence level of 95% calculated by the equation (Martin, 2000; IAEA, 1989):

$$\mu C = 2C \sqrt{\left(\frac{\sigma Nt}{Nt}\right)^2 + \left(\frac{\sigma Nb}{Nb}\right)^2 + \left(\frac{\sigma \varepsilon}{\varepsilon}\right)^2 + \left(\frac{\sigma P\gamma}{P\gamma}\right)^2 + \left(\frac{\sigma Fk}{Fk}\right)^2}$$

Where σNt, σNb, σε, σPγ and σFk were the standard deviation of each sample count rate, background count rate, efficiency, gamma radiation and self-absorption correction factor respectively.

Materials and Methods

a. Gamma Radiation Exposure Rate Measurements and Surface Contamination

Gamma radiation exposure rate of TENORM was measured using surveimeter Micro R Meter Model 19 with detector NaI (Tl) made by Ludlum Measurement Inc. USA. This tool is designed to measure natural gamma radiation exposure at a rate of 1 micro Roentgen per hour up to 5000 micro rontgent per hour (1-5000 μR/h) with a standard deviation of ± 10%. Gamma exposure rate was calculated using equation 1. Each work area that has a gamma radiation exposure rate of more than 50 μR/h was marked as a Hot Spot. TENORM surface contamination was measured using a Mini Monitor Series 900 with probe type scintillation detector made Mini Instrument Ltd. UK. Detector probe is placed at a distance of 5 mm from the surface of TENORM. Surface contamination is calculated using equation 2.

b. Measurement of Radionuclide Activity Concentrations TENORM

TENORM which was sampled of measurement points in the offshore oil and gas industry A, B and C, dried at a temperature of 110^{o}C for at least 24 hours to be free from water vapor. The dried samples were crushed using a grinder or mortar and sieved with a 100 mesh sieve diameter, homogenized and put into a marinelli beaker with volume of one liter and then sealed using glue araldit and stored for one month to get go to for equilibrium between ^{226}Ra and ^{228}Th with decay daughter.

HPGe Gamma Spectrometer made by ORTEC models GEM-60 besetted at working voltage, coarsa gain, fine gain and amplifier according to manuals. Standard source with volume of one liter placed in a container with a density of 1.00 gr/cm^{3}marinelli containing multiple isotopes of ^{54}Mn, ^{60}Co, ^{65}Zn, ^{109}Cd, ^{133}Ba, ^{134}Cs, ^{137}Cs, ^{210}Pb dan ^{241}Am with each activity: (152.87 ± 0, 94; 225.06 ± 2.21; 400.49 ± 4.19; 346.51 ± 2.76; 174 ± 1.28; 51.67 ± 0.52; 62.77 ± 0.42; 405, 15 ± 2.54 and 294.47 ± 2.00) Bq was made in PTKMR-BATAN on October 1, 2004 are traceable to the Laboratory of IAEA. Life time artificial ORTEC HPGe Gamma Spectrometer Model GEM-60 was set in interval of 3600 seconds, then do the enumeration.

Peak energy generated from standard sources enumeration of multi isotopes of ^{54}Mn, ^{60}Co, ^{65}Zn, ^{109}Cd, ^{133}Ba, ^{134}Cs, ^{137}Cs, ^{210}Pb dan ^{241}Am were used to change the stripe number on gamma spectrometer. An example is the stripe number on peak energy of gamma spectrometer replaced ^{54}Mn (acquisition) with peak energy ^{54}Mn and so on for the other peak energy in order to obtain a range from the lowest energy range (81.00 keV energy for ^{133}Ba) up to the highest energy (1332.50 keV) for ^{60}Co calibration energy.

Standard enumeration of the source is then calculated for counting efficiency of each peak (peak) using equations 3 and 4. Enumeration efficiency calculation results from each source of energy in the standard was subsequently graphed for the relationship with energy efficiency in order to obtain the efficiency of the calibration curve used to determine the efficiency of the other radionuclides that were not known.

Furthermore marinelli beaker containing TENORM sample was placed on the detector HPGe Gamma Spectrometer then counted for 3600 seconds. Activity concentration of radionuclides in TENORM sample was calculated using equation 5 and 6.

Results and Discussion

Results of gamma radiation exposure rate sand surface contamination on some offshore oil and gas industries in Java are presented in Table2. Gamma radiation exposure rate ranged from not detected to 38.0 ± 2.8 μR/h, 42.5 ± 4.9 μR/h and 135.0 ± 7.1 μR/h. Value of the rate of radiation exposure at some measurement point is above the background exposure

rate but still below the maximum permissible limit of 50 μR/h, but there is one point of measurement that exceeds the maximum allowed in the production area of the header line (1) on C oil and gas industry with a value of 135.0 ± 7.1 μR/h. The rate of radiation exposure with a value ≥ 50 μR/h marked as a hot spot (API, 1992).

Table 2. Gamma radiation exposure rate and surface contamination on the several offshore oil and gas industries in Java Island.

No	Location	Gamma radiation exposure rate (μR/h)	Surface contamination(Bq/cm^2)
Oil and Gas Industry A			
1	Compessor-1 Sytem P/F :		
	All upperdeck area	n.d	n.d
	Slug catcher V-23-01	14.0 ± 1.4	n.d
	All Cellar deck area	n.d	n.d
2	ProccesSystemP/F:		
	Atmospheric separator DV-5	17.0 ± 4.2	n.d
	Knock drum DV-8 bottom	34.5 ± 0.7	1.26 ± 0.4
	All cellar deck area	n.d	n.d
	All landing boat deck area	n.d	n.d
3	Compressor -2SytemP/F :		
	V-2A vessel	19.5 ± 0.7	n.d
	PCV-4-2B	9.0 ± 1.4	n.d
	V-3 vessel	17.5 ± 3.5	n.d
	V-6A vessel bottom	38.0 ± 2.8	1.88 ± 0.13
	V-6B vessel bottom	35.0 ± 2.8	1.88 ± 0.13
	V-5A vessel bottom	9.0 ± 1.4	0.83 ± 0.10
	V-4 srubber bottom	19.5 ± 2.1	0.53 ± 0.04
	V-5B vessel	8.5 ± 0.7	n.d
	V-2B vessel bottom	22.5 ± 0.7	1.88 ± 0.30
	Gotter main deck (near V-400 vessel)	22.5 ± 0.07	1.67 ± 0,53
	Valve bypass inlet to V-400 12”	29.0 ± 1.4	n.d
	All cellar deck area	n.d	n.d
	All landing boat area	n.d	n.d
4	WellSystem P/F :		
	All cellar deck area	n.d	n.d
	All main deck area	n.d	n.d
5	Service SytemP/F :		
	Generator set area	7.0 ± 1.2	n.d
	All cellar deck area	n.d	n.d
	All main deck areaa	n.d	n.d
Oil and Gas Industry B			
1	Process System P/F		
	Floor inside wall turbin room (East)	42.5 ± 4,9	0.25 ± 0.02
	Floor inside wall turbin room (West)	14.0 ± 1,4	0.06 ± 0.01
	All upper deck area		
	Bottom FP-V-4HPF-Ko drum	17.5 ± 0.7	1.30 ± 0.09
	Bottom V-7 Ko drum	19.9 ± 0.7	0.50 ± 0.04
	Bottom Diesel Machine	11.0 ± 1.4	0.15 ± 0.01
	Floor panel control	6.0 ± 2.8	0.24 ± 0.02
	Gotter near FP-V-4HPF Ko drum	6.0 ± 0.7	0.08 ± 0.01
	Sand (near paint storage)	11.0 ± 0.7	0.13 ± 0.01
	All landing boat deck area		
2	Service System P/F		
	All cellar deck area	n.d	n.d
	All main deck area	n.d	n.d
Oil and Gas Industry C			
1	Compressor System P/F :		
	Path separator control room and	23.0 ± 1.4	0.42 ± 0.06

	compressor		
	All upper deck area	n.d	n.d
	05-V-40 tank inside	21.5 ± 0.7	n.d
	05-V-40 tank bottom	25.0 ± 1.4	0.73 ± 0.05
2	Procces SytemP/F :		
	LPV-6 separator tank	13.5 ± 0.7	n.d
	All upper deck area	n.d	n.d
	Diesel engine bottom	13.0 ± 1.4	n.d
3	Well Sytem P/F :		
	All main deck area	n.d	n.d
	Production header line area (1)	135.0 ±7.1	n.d
	Production header line area (2)	34.0 ± 5.7	n.d
	Grating floor (inside instrument panel)	6.5 ± 0.7	0.63 ± 0.04
4	Service System P/F :		
	Floornear ESD panel	5.5 ± 0.7	0.74 ± 0.05
	All main deck area	n.d	n.d
	All upper deck area	n.d	n.d
	All top deck area	n.d	n.d
	All helipad deck area	n.d	n.d

Notes :n.d : not detectable for gamma exposure rate were ≤ 2.5 ± 0.5 μR/hour and surface contamination were ≤ 0.01 ± 0.001 Bq/cm^2.

The high radiation exposure rate in the production header line area (1) on C oil and gas industry because TENORM deposited inside the operating system in the form of scale and/or sludge. This is because the equipment had already been used for long enough time although now no longer used. Duration of employment arrangements need to be done so that workers who work in these places received dose order Dose Limit Value (NBD) as the general public that is equal to 1 mSv/year (Perka BAPETEN No. 4, 2013). In order that the NBD is not exceeded worker should not working at the site for 714.29 hours in one year. In addition to setting the duration of the work to be done, in the production line header area (1) need to be installed/placed a warning sign of radiation exposure to TENORM at 135.0 ± 7.1 μR/h so that unauthorized persons are in that place.

Contamination on the surface of the oil and gas industry A, B and C were between not detected up to 1.88 ± 0.13 Bq/cm^2, 1.30 ± 0.09 Bq/cm^2 and 0.74 ± 0.05 Bq/cm^2 respectively. Value of surface contamination at some point measurements has exceeded the permitted limit value of 0.4 Bq/cm^2 so that it necessary o tae remedial action (Perka BAPETEN No. 4, 2013). Remedial actions are actions that restore normal radioactive concentrations to be below the intervention level of dose levels that can be avoided with protective or remedial actions to situations of chronic exposure or emergency exposure (Perka BAPETEN No. 9, 2009). Value measured of surface contamination is TENORM contaminants that are outside the operating system, while TENORM inside operation systems (which are in the pipeline) is not measurable due to the pipe walls.

It also need to be aware that the TENORM deposited inside surface of the production system or in other operating systems that are currently not detected any indication of such TENORM were in: tubular, wellhead, valves, pump, separator and others places. TENORM is usually deposited in the form of thin films or coatings in the form of a black powder that commonly referred to black powder. In Black powder usually contain ^{210}Pb and ^{210}Po radioactive substances that emit alpha radiation and heavy metals such as arsenic (As) and mercury (Hg). Both radioactive substances and heavy metals that are very dangerous if be entrered the human body. The hazardous substance can enter the human body through the oral route, and skin or inhalation/breathing. Therefore when the production system is opened for maintenance or cleaning, the workers must use appropriate PPE.

Concentration of radionuclides of ^{232}Th, ^{228}Th, ^{226}Ra and ^{40}K of soil/sand sample from the oil and gas industry A are presented in Table 3. ^{232}Th and ^{228}Th concentrations were ranged from (3,504 ± 330) Bq/kg and (3644 ± 187) Bq/kg in drum knock up DV-8 bottom (18,450 ± 1,766) Bq/kg and (18 291 ± 1738) Bq/kg in V-4 Scrubber bottom. ^{226}Ra concentrations were ranged between (1,450 ± 137) Bq/kg in bottom drum knock up DV-8 (9501 ± 894) Bq/kg in gotter Main Deck (near V-400 vessel) and ^{40}K concentrations were ranged between (353 ± 36) Bq/kg in bottom DV-8 drum knock up (1,545 ± 219) Bq/kg in bottom V-4 scrubber.

According to Regulation of BAPETEN No. 9 Year of 2009 on the intervention of TENORM exposure, the concentration of ^{232}Th, ^{226}Ra and ^{228}Th in six points are above screenshot the maximum allowed limit (1000 Bq/kg) an intervention should be carried out. In this research the concentrations of ^{40}K is below the maximum allowable limits so there is no

need intervention. However, because the concentration of radionuclides ^{40}K mixed sample in the soil/sand are above the maximum allowable limits, then the entire sample should be done interventions.

Concentration of ^{232}Th, ^{228}Th, ^{226}Ra and ^{40}K radionuclides in sand/soil and sludge samples were sampled from the oil and gas industry B are presented in Table 4. Concentration of ^{232}Th and ^{228}Th respectively were ranged from 50 ± 5 Bq/kg and 35 ± 3 Bq/kg in sludge tanks to collect, 10,983 ± 1,035 Bq/kg and 11.644 ± 809 Bq/kg in bottom Diesel Machine. While the concentration of ^{226}Ra and ^{40}K respectively were ranged from 93 ± 10 Bq kg and 43 ± 6 Bq/kg in Gotter near FP-V-4 HPF Ko drum to 4.625 ± 438 Bq/kg and 860 ± 88 Bq kg. TENORM are found in the turbine room floor inside wall (East), bottom FP-V-4 Drum HPF-Ko, Ko bottom V-7 and in the bottom drum machine on industrial diesel oil B has exceeded the maximum allowed limit, so it must be taken intervention. While TENORM are found in gotter near FP-V-4 HPF Ko drums, tanks and sludge collect sand near paint storage is still below the maximum allowable limits.

Concentration of ^{232}Th, ^{228}Th, ^{226}Ra and ^{40}K radionuclides of scale, soil and sludge samples in the oil and gas industry C are presented in Table 5. ^{232}Th concentrations were ranged between (0.14 ± 0.02) Bq/kg up to (7,400 ± 696) Bq/kg, ^{228}Th from (0.12 ± 0.01) Bq kg up to (7,342 ± 493) Bq/kg, the concentration of ^{226}Ra were ranging between (0.06 ± 0.01) in Bq/kg up to (2,806 ± 265) Bq/kg and concentrations ranged ^{40}K (0.09 ± 0.01) Bq/kg up to (514±51) Bq/kg. The lowest TENORM concentration value in the alley way and the highest in the 05-V-40 tank bottom. The concentration of ^{232}Th, and ^{226}Ra in Grating ^{228}Th floor inside the instrument panel and in the 05-V-40 tank bottom is above the maximum allowable limit, so need an intervention. Whereas the concentrations of TENORM in Path separator was below the maximum allowable limits so no need for intervention.

Table 3. Radionuclide concentrations sample soil or sand in oil and gas industry A.

No.	Location	TENORM	Concentration TENORM (Bq/kg)			
			^{232}Th	^{228}Th	^{226}Ra	^{40}K
1.	Bottom Knock drum DV-8	Soil/sand	3,504± 330	3,644± 187	1,450± 137	353± 36
2	Gotter Main Deck	Soil/sand	6,271 ± 590	6,764 ± 636	9,507± 894	517± 52
3	Bottom V-2B vessel	Soil/sand	15,977 ± 1,504	15,015 ± 1,412	6,364± 601	840 ± 88
4	Bottom V-6A vessel	Soil/sand	7,442± 700	7,939 ± 401	3,187± 301	566± 57
5	Bottom V-6B vessel	Soil/sand	8,168± 768	8,681± 589	3,348± 316	558 ± 56
6	Bottom V-4 Scrubber	Soil/sand	18,450 ± 1,766	18,291 ± 1,738	7,808 ± 766	1,545± 219

Tabel 4. Radionuclide concentrations sample soil/sand and sludge in oil and gas industry B.

No.	Location	TENORM	Concentration TENORM (Bq/kg)			
			^{232}Th	^{228}Th	^{226}Ra	^{40}K
1.	Floor inside wall turbin room (East)	Sand/Soil	1,745± 165	1,870 ± 126	748± 71	227± 24
2	Bottom FP-V-4 HPF-Ko Drum	Sand/Soil	6,560 ± 619	7,022 ± 477	2,889± 274	564± 59
3	Bottom V-7 Ko drum	Sand/Soil	3,314± 314	3,021± 194	1,415 ± 136	304 ± 34
4	Gotter near FP-V-4 HPF Ko drum	Sand/Soil	207± 20	212± 14	93±10	43 ±6
5	Sludge collect tank	Sludge	50 ± 5	35 ±3	253± 24	385± 37
6	Bottom Diesel machine	Sand/Soil	10,983 ± 1,035	11,644 ± 809	4,625± 438	860 ± 88
7	Sand near paint storage	Sand	278± 26	259 ± 17	266 ± 25	119 ± 12

Table 5. Radionuclide concentrations sample soil/sand, scale and sludge in oil and gas industry C

No.	Location	TENORM	Concentration TENORM (Bq/kg)			
			^{232}Th	^{228}Th	^{226}Ra	^{40}K
1.	05-V-40 tank bottom	Sand/Scale/Soil	7,400 ± 696	7,342 ± 493	2,806 ± 265	514 ± 51
2	Grating floor inside instrument panel	Sludge/Soil	2,032 ± 193	1,956 ± 185	857 ± 82	225 ± 26
3	Path separator	Scale/Soil	0.14 ± 0.02	0.12 ± 0.01	0.06 ± 0.01	0.09 ± 0.01

Intervention action in offshore oil and gas industry is done by cleaning TENORM in the area remaining operation, collected, put in a plastic bag and then sealed drums. The next drum marked TENORM danger, put in place that rarely work activities are like in the landing boat and tied so as not to fall into the sea when the wind blown.

TENORM are found in the oil and gas industry A, B and C are generally derived from waste material sandblasting work. Therefore you should avoid the use of sandblasting materials containing radioactive substances such as copper slag and tin slag, and selected material that is not radioactive like silica or garnet. If forced to use materials containing radioactive substances, then after the use of waste material must be managed in a way collected and stored safely before wasted.

Conclusion

The radiation exposure rate in the oil and gas industry A, B and C was respectively ranged from not detected up to 38.0 ± 2.8 μR/h, 42.5 ± 4.9 μR/h and 135.0 ± 7.1 μR/h, while the surface contamination was ranged from not detected to 1.88 ± 0.30 Bq/cm^2, 1.3 ± 0.09 Bq/cm^{2}and 0.74 ± 0.05 Bq/cm^2. The type and level of radionuclide concentrations were ranged from 0.14 ± 0.02 Bq/kg to 18,450 ± 1,766 Bq/kg for ^{232}Th radionuclide, between 0.12 ± 0.01 Bq/kg up to 18 291 ± 1738 Bq/kg for ^{228}Th radionuclide, between 0.06 ± 0.01 Bq/kg up to 9507 ± 894 Bq/kg for ^{226}Ra radionuclide and between 0.09 ± 0.01 Bq/kg up to 1,545 ± 218 Bq/kg for ^{40}K radionuclide. Under the term BAPETEN regulation the value of radiation exposure, contamination of surface and radionuclide concentration levels at some point of measurement of TENORM in several oil and gas industry on the island of Java, has exceeded the allowed limit value so that it is necessary to remedial action and a restriction in workers' employment worktime in order to control as value the general public dose limit is not exceeded.

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DISCUSSION

1. I Gde Sukadana,

   Question : Why the concentrations of TENORM that were taken from oil and gas industries did not compared to the TENORM from around the area?

   Answer : Concentration of TENORM in industries under the study did not necessary to be compared to the value of TENORM in around the industries because we only intended to know/determine TENORM in that area, while there is no TENORM found in around the area (it was clean).

2. Dr. Abel J. Gonzalez, ARN

   Question : The highest concentration of TENORM in oil and gas industries usually found inside the pipes, but why did in your research the highest was found in other places (deck). Please explain.

   Answer : Concentration of TENORM was found not in the pipes because that TENORM was mostly came from the rest of sandblasting activities (types of cooper slag and tin slag), whereas TENORM with the type of scale and sludge from pipes were found in low amount.

Verification of the Efficiency of HPGe Semiconductor Detector Using Standard Source of PTKMR BATAN Product

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Abstract. **Center for Technology of Radiation Safety and Metrology (PTKMR) - National Nuclear Energy Agency (BATAN), has been conducted a research and development on the manufacture of standard sources used for calibration of radiation instruments. One of the counter systems for the measurement of radionuclide activity is a gamma spectrometer. Radioactivity measurement with this method by means of quantitative analysis by calibration efficiency. Full energy peak efficiency in a gamma spectrometer is absolute efficiency of the total absorption peak. Total absorption peak is the number of counts contained in a radionuclide for gamma energy peaks. The aim of this research was to verify the efficiency of High Purity Germanium (HPGe) semiconductor detector using standard sources of PTKMR BATAN standardization laboratory product. In this research, PTKMR BATAN standard source used were: ^{152}Eu (T1/2: 13.522 years), ^{60}Co (T1/2: 5.23 years), ^{137}Cs (T1/2: 30.05 years) and ^{133}Ba (T1/2: 10.54 years) and Laboratoire Metrologie des Rayonnements Ionisants (LMRI) French standard source in the 121 keV-1408keV energy range for measuring point source geometry with the source to detector distance of 25 cm. From the analysis of HPGe semiconductor detector efficiency calibration curve using PTKMR standard sources and LMRI standard sources shows that differences in the efficiency of the energy of each gamma was quite good about 4%.**

Keywords: Verification, efficiency calibration , gamma spectrometer, standard source

Introduction

Gamma spectrometer counting system is one means of measuring the relative radioactivity measurement. Radionuclide activity is determined by comparing the results of sample counting activity with radionuclide standard sources. Method of radioactivity measurement using gamma spectrometer counting system called gamma spectrometry. Analysis used in the gamma spectrometry method is based on the interpretation of gamma spectrum measurements. The interaction between radionuclide gamma rays with detector produces pulses of comparable radionuclide gamma energy and ultimately processed electronically resulting gamma spectrum. Prior to the radioactivity measurement using gamma spectrometer counting system, the counting system must be calibrated because precision and accuration of measurement equipment is depending on the condition of the equipment. Several steps that need to be done are to determine the efficiency of the counting system using radionuclide standard source geometry which includes sample shape, sample size, and detector and source distance.

Quantitative analysis of the gamma spectrometer counting system is done by efficiency calibration. The efficiency calibration is based on the notion that essentially any radionuclide always emit radioactive rays in all directions. The measurement radioactivity performed at a certain distance from the detector. So the detector efficiency to radionuclide gamma rays known can result in part of the emitted gamma rays. The result of the gamma spectrometer efficiency calibration is curve between efficiency and gamma energy. Efficiency calibration curve was prepared by measuring and calculating the efficiency of a radionuclide that has a gamma energy of low to high energy range, typically using standard sources such as ^{152}Eu, ^{133}Ba, ^{137}Cs and ^{60}Co standard source or mixed standard source (NCRP, 1978). The quality of the efficiency of calibration measurement results affect so precision and accuracy in manufacturing efficiency calibration curve is very important. Precision and accuracy of making the efficiency calibration curve is very dependent on the determination of the peak area absorption total each gamma ray energy spectrum. Determination of the spectral peak area will determine the counting rate, count per second (cps). In addition it is also determined by emission rate which is the multiplication of the intensity of the radionuclide activity at measurement time.

The aim of this research is to verify the efficiency of *High Purity Germanium* (HPGe) semiconductor detector using standard sources of PTKMR BATAN standardization laboratory product. In this research, PTKMR BATAN standard source used are: ^{152}Eu, ^{60}Co ^{137}Cs and ^{133}Ba and *Laboratoire Metrologie des Rayonnements Ionisants*

(LMRI) French standard source in the 121 keV-1408keV energy range for measuring point source geometry with the source to detector distance of 25 cm.

Multi gamma energy of radionuclide of ^{152}Eu has gamma energy range from 121 to 1408 keV, the 121.8 keV with intensity (28.37%), 244.6974 keV (7.55%), 344.2785keV (26.59%), 411.1 keV (2.23%), 443.965 keV (2.8%), 778.9045 keV (12.97%), 964.079 keV (14.5%), 1112.076 keV (13.41%) and 1408.013 keV (20.85%). ^{133}Ba has gamma energy 356 keV (62.05%). ^{137}Cs has gamma energy 661.66 keV (84.99%) and ^{60}Co has gamma energy 1173.23 keV (99.85%), 1332.49 keV(99.98%). The use of radionuclide standard source is very useful and efficient for efficiency calibration radionuclide measurement in the range of low to high gamma energy can be carried out simultaneously or one measurement so it saves measurement time.

In the radioactivity measurement of using a radioactive source with gamma spectrometer counting system there are several factors that need to be considered, such as the type of detector used, detector efficiency and detector resolution. Detector efficiency is a measure of how many pulses occur for a given number of gamma rays. Various kinds of efficiency definitions are in common use for gamma ray detector, absolute efficiency and intrinsic efficiency. Absolute efficiency is the ratio of the number of counts produced by the detector to the number of gamma rays emitted by the source. At this efficiency depends on the geometry counting is the distance between the radioactive source and the detector. While the intrinsic efficiency is the ratio of the number of pulses produced by the detector to the number of gamma rays striking the detector. It just depends on the efficiency of the detector material and radiation energy.

At HPGe detector gamma spectrometer the counter system that is efficiency often used is the absolute efficiency. The efficiency values according to the equation:

$$\varepsilon(E) = \frac{cps}{dps \times Y(E)} \quad \text{.....(1)}$$

where:
ε (E) is the absolute efficiency of the gamma energy, cps is the count rate measurements dps is a standard activity performed at the time of measurement (Bq), Y (E) is the intensity or yield a radionuclide as a function of gamma energy.

The detection efficiency value of measurements are generally determined by geometry factors such as the source to detector distance, sample shape and size, active volume of the detector, detector resolution. If the distance of the radionuclide source close to the detector, the greater of the detection efficiency value. At this distance, the greater the amount of measurement error due to the effects and symptoms of pile-up. Moreover the form of the radionuclide source geometry will influence the measurement results. Measuring the distance between the detectors on the same sources of radionuclide, the radionuclide source form will be shaped different point values with the efficiency of radionuclide sources planset shaped. The detector has an active volume of the well produce different efficiency values with small active volume. The larger the detector active volume, the greater the value of efficiency and increasing the cost.

Materials and Methods

Preparation of ^{152}Eu, ^{133}Ba, ^{137}Cs and ^{60}Co

The original ^{152}Eu source in the form of Europium (III) chloride, EuCl3 in 1 M HCl, solution, ^{137}Cs source in the form of Cesium chloride, CsCl in 0.1 M HCl solution, ^{133}Ba source in the form of Barium chloride $BaCl_2$ in 0.1 M HCl solution and ^{60}Co source in the form of Cobalt (III) chloride, $CoCl_2$ in 0.1 M HCl solution provided by POLATOM – Polandia were prapared in the point source. Solutions were prepared into solid point source sucked into a small polyethylene ampoule (baby bottle), and then dropped onto a thin polyester film as source support is about from 5,30 to 20,54 mg. The small polyethylene ampoule were carefully weighed in each case to determine the mass of the drops. And then, the sources were dried in a chamber with circulating air. After dry, the sources were covered with thin polyester film.

Measurement of sample

Activity concentration of ^{152}Eu, ^{133}Ba, ^{137}Cs and ^{60}Co as solid point source sample were measured twice replicates for each sample. The activities are measured using gamma-spectrometer system HPGe detector, with source and detector distance of 25 cm. Efficiency calibration carried out using Eu-152 standard source of the LMRI – France.

Results and Discussion

Radioactivity measurements of ^{152}Eu standard source LMRI used to calculate the efficiency of each gamma energy ^{152}Eu for measuring point source geometry with the source to detector distance of 25 cm with HPGe detector. By using equation (1) obtained the value of efficiency (ε) detection High Purity Germanium detector (HPGe) using ^{152}Eu stan dard source at each gamma energy. Value measurement efficiency (ε) is presented in Table 1.

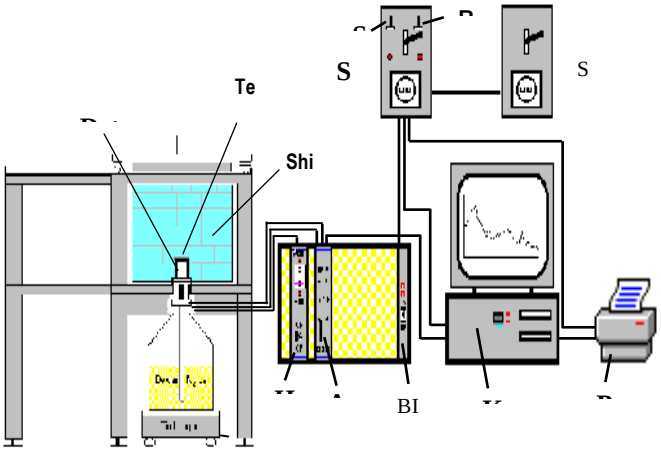


Figure 2. Scheme of the gamma spectrometer system.

Of the value of the efficiency of the calibration curve can be made as a function of gamma energy efficiency at each distance measurement. Efficiency calibration curve is presented in Figure 1

Standard source PTKMR BATAN ^{152}Eu (T1/2: 13.522 years), ^{60}Co (T1/2: 5.23 years), ^{137}Cs (T1/2: 30.05 years) and ^{133}Ba (T1/2: 10.54 years) was made in some periods. The standard sources should be verified with the standard source LMRI. The purpose of this measurement is to check the accuracy of the measurement results by comparing the efficiency value of PTKMR radionuclide and Laboratoire Metrologie des Rayonnements Ionisants (LMRI) French standard. LMRI standard source used was ^{152}Eu multi gamma radionuclide.

According to Debertin (1985) at energies below 300 keV the possibility of summing effects. Difference in the efficiency value of PTKMR standards source compared with standard sources LMRI of 0 - 5.67%. In the efficiency calibration curve in the region above 300 keV gamma energy, differences of efficiency value betwen PTKMR standard source and LMRI standard sources quite well under 4%. While under 300keV gamma energy, differences of efficiency value betwen PTKMR standard source and LMRI standard sources above 5%. This is because the region of low energy gamma the ability to interact with gamma rays detector very low. So the ability to penetrate the active window detector is also lower. In contrast with the increasing of energy gamma where gamma photons escape the detector without interacting be large enough so that the efficiency of detection would also go down.

Table 1. High Purity Germanium Detector Efficiency Value with PTKMR standard source product and LMRI; (2009: ^{152}Eu, ^{60}Co, ^{133}Ba, ^{137}Cs).

Source	Energy (keV)	Yield	Efficiency (PTKMR)	Efficiency (LMRI)	%
Eu-152	121,8	0,2837	0,00113	0,00120	5,67
Eu-152	344,3	0,2658	0,00050	0,00048	-3,79
Eu-152	778,9	0,1296	0,00025	0,00024	-3,12
Eu-152	964	0,1462	0,00020	0,00020	-3,03
Eu-152	1408,1	0,2085	0,00015	0,00014	-3,42
Ba-133	356	0,621	0,00049	0,00047	-3,85
Co-60	1173	0,999	0,00017	0,00017	-2,82
Co-60	1332	0,999	0,00015	0,00015	-3,63
Cs-137	661,8	0,899	0,00028	0,00027	-1,75

Table 2. High Purity Germanium Detector Efficiency Value with PTKMR standard source product and LMRI; (2010: ^{60}Co, ^{133}Ba, ^{137}Cs).

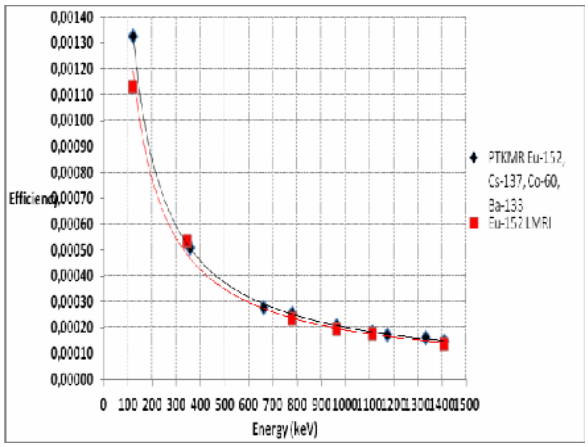
Source	Energy (keV)	Yield	Efficiency (PTKMR)	Efficiency (LMRI)	%
Co-60	1173,23	0,9985	0,000156	0,000158	1,15
Co-60	1332,49	0,99983	0,000138	0,000139	0,70
Ba-133	356,00	0,6205	0,000504	0,000507	0,51
Cs-137	661,66	0,8499	0,000274	0,000276	0,72

Table 3. High Purity Germanium Detector Efficiency Value with PTKMR standard source product and LMRI; (2011: ^{152}Eu, ^{60}Co, ^{137}Cs).

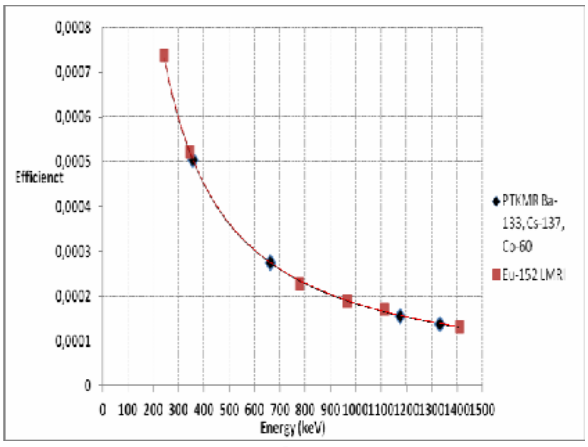
Source	Energy(keV)	Yield	Efficiency (PTKMR)	Efficiency (LMRI)	%
Co-60	1173,23	0,9985	0,000154	0,000155	0,04
Co-60	1332,49	0,999826	0,000137	0,000137	-0,52
Cs-137	661,66	0,8499	0,000274	0,000269	-1,76
Eu-152	244,6974	0,0755	0,000739	0,000708	-4,41
Eu-152	344,2785	0,2659	0,000507	0,000508	0,32
Eu-152	778,9045	0,1297	0,000225	0,000230	2,14
Eu-152	964,079	0,145	0,000182	0,000187	2,54
Eu-152	1112,076	0,1341	0,000165	0,000163	-1,18
Eu-152	1408,013	0,2085	0,000130	0,000129	-0,72

Table 4. High Purity Germanium Detector Efficiency Value with PTKMR standard source product and LMRI; (2012: ^{60}Co, ^{133}Ba, ^{137}Cs).

Source	Energy(keV)	Yield	Efficiency (PTKMR)	Efficiency (LMRI)	%
Co-60	1173,23	0,9985	0,000163	0,000156	-4,63
Co-60	1332,49	0,999826	0,000131	0,000137	4,63
Ba-133	356,00	0,6205	0,000506	0,000506	0,00
Cs-137	661,66	0,8499	0,000274	0,000274	0,00



(a)



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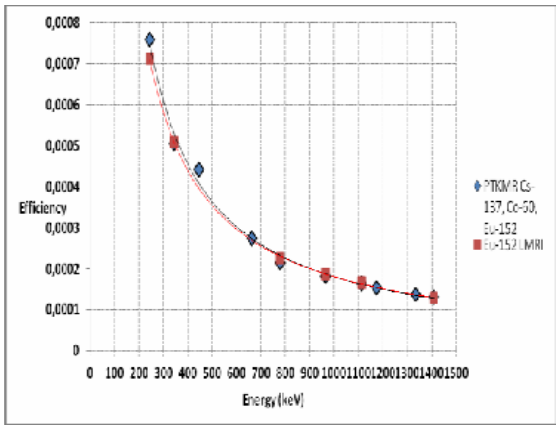

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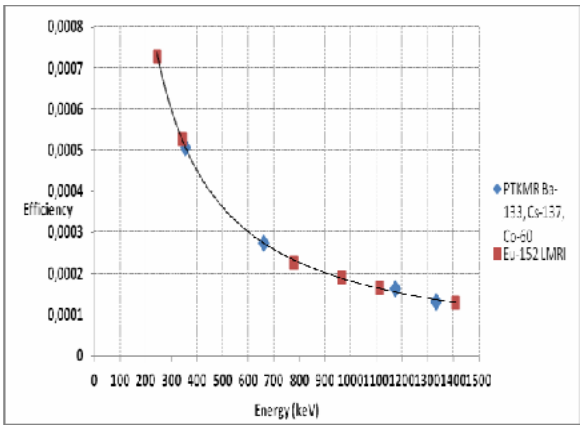

(d)

Figure 1. High Purity Germanium Detector Efficiency Curve with standard source PTKMR product and LMRI.

Conclusion

From the research that has been done there are somethings that can be concluded that the quantitative analysis of, efficiency calibration by using gamma spectrometry is a method that has a high accuracy. Verification of the efficiency value with *High Purity Germanium* (HPGe) semiconductor detector using standard sources of PTKMR BATAN standardization laboratory product using PTKMR BATAN standard source : ^{152}Eu (T1/2: 13.522 years), ^{60}Co (T1/2: 5.23 years), ^{137}Cs (T1/2: 30.05 years) and ^{133}Ba (T1/2: 10.54 years) and Laboratoire Metrologie des Rayonnements Ionisants (LMRI) French standard source in the 121 keV-1408keV energy range for measuring point source geometry with the source to detector distance of 25 cm. From the analysis of HPGe semiconductor detector efficiency calibration curve using PTKMR standard sources and LMRI standard sources shows that differences in the efficiency of the energy of each gamma is quite good about 4%.

Acknowledgements

Author thankful for the opportunity by Research Group for Standardization of Radionuclides, Center for Technology of Radiation Safety and Metrology BATAN facilities that provide measurements for the research.

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Calibration of Ionization Chamber of IG11/A20 in PTKMR BATAN

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Abstract. **Calibration of ionization chamber Centronic of IG11/A20 as a secondary standard measuring system at the Center for Technology of Radiation Safety and Metrology (PTKMR), National Nuclear Energy Agency (BATAN) was performed. Efficiency calibration has been carried out with and without Fe absorber to minimize the difficulties in the measurement of low energy region. Efficiency calibration has been conducted without and with absorber with thickness of 1 mm for energy below 250 keV. The consistency of the calibration curve was checked by measuring and calculating efficiency value of without and with absorber for ^{134}Cs and ^{133}Ba. The result was relatively good with the difference about 0.4% to 0.6% for with and without absorber, respectively.**

Key words : ionization chamber, absorber, efficiency curve

Introduction

The re-entrant or 4πγ ionization chamber Centronic IG11/A20 system was developed as a secondary standard of radionuclide calibrator at the PTKMR BATAN. Ionization chamber measuring systems are widely used for activity measurement of photon-emitting radionuclides and to assure the quality of standards disseminated to user (Schrader, 2000; Schrader, 2002). A re-entrant or 4πγ ionization chamber can be calibrated with appropriate radioactive standard sources in defined geometry that were standardized by primary method (Schrader, 1997). For the calibration, nine Physikalisch-Technische Bundesanstalt (PTB) activity standards in flame sealed ampoules with about 2 g weight of solution and three PTKMR activity standards were used.

The aim of this research was to improve the ability of PTKMR-BATAN in the field of radiation metrology especially in standardization of radionuclides and make Centronic IG11/A20 Ionization chamber 4 π-γ as a secondary standard instrument for the measurement of activity.

Materials and Methods

The Ionization chamber coupled with an electronic system consists of Electrometer Model 642 Keithley, Electrometer Remote Head Model 642 Keithley, High Voltage Supply Model 248 Keithley. In this study the energy-dependent efficiency curve are discussed.

The ionization chamber current I originating from a source is proportional to the activity *A* of the measured sample. The ^{226}Ra reference source also applies to the current I_{Ra} . The proportionality factor is the radionuclide efficiency ε_N by the relation (Svec. A., Schrader, 1997, 2000), :

$$\frac{I}{I_{Ra}} = \varepsilon_N A \qquad (1)$$

In the current measuring instrument, it is possible to adjust the instrument reading by an internal and eksternal instrument setting. An Ionization chamber the current component are detected and collected and their sum is the total ionization current measured. The relative nuclide efficiency ε_N can be expressed as the sum (Svec. A., Schrader, H., 1997, 2000):

$$\varepsilon_N = \sum p(Ei)\varepsilon(Ei) \qquad (2)$$

Where *p(Ei)* is the probability for emission per decay of a photon with energy *Ei* and *ε(Ei)* the relative photon efficiency at this energy.

Twelve radionuclide standards were measured for calibration, see table 1. The calibration procedure was used in accordance with the principle of the method described in detail by Schrader (1997, 2000). The following PTB standard sources used for calibration were ^{241}Am, ^{210}Pb, ^{54}Mn, ^{60}Co, ^{137}Cs, ^{22}Na, ^{65}Zn, ^{109}Cd and ^{139}Ce and also PTKMR standard source ^{57}Co, ^{51}Cr and ^{58}Co. The measured ionization currents were corrected for background and decay during experiment. To determine the efficiency curve, it was started with practically monoenergetic radionuclides with energies in the linier region of the curve such as e.g. ^{54}Mn, ^{51}Cr and ^{60}Co (Schrader, 1997). The long-term stability is checked routinely every month by using ^{226}Ra National Bureau of Standards (NBS) reference source. Efficiency calibration was conducted without and with absorber Fe. With absorber of thickness of 1 mm at energies below 250 keV a significant decline in efficiency.

Table 1. Radionuclide standards were measured for calibration of ionization chamber.

Nuclide	*E* (keV)	*p*	Remark
^{210}Pb	46.5	0,042	PTB standard source
^{241}Am	59.5	0,360	PTB standard source
^{109}Cd	88.03	0,003	PTB standard source
^{139}Ce	165.86	0,800	PTB standard source
^{22}Na	511	1,798	PTB standard source
^{137}Cs	661.66	0,850	PTB standard source
^{54}Mn	834.84	1,000	PTB standard source
^{65}Zn	1115.5	0,504	PTB standard source
^{60}Co	1252.9	1,999	PTB standard source
^{57}Co	123.6	0,962	PTKMR standard source
^{51}Cr	320	0,098	PTKMR standard source
^{58}Co	810.8	0,995	PTKMR standard source

Results and discussion

The efficiency curve of ionization chamber without and with absorber is plotted in Figure 1.

Figure 1 shows that there is an almost constant increase at energies above 250 keV, where the Compton effect dominates, while the peak at lower energies is due to the rapid increase in photoelectric effect with decreasing energy and to the attenuation of low-energy photon (Svec. A., Schrader, H., 2002).

Moreover, it proves particularly useful to measure all radionuclides with a Fe-liner of 1 mm thickness around the sources. An absorber of this thickness discriminate against low-energy photon and considerably reduces the peak of the efficiency curve, while at higher energies, it alters the efficiency only slightly. From both measurements, with and without the Fe-absorber, several efficiency points in the low energy region could be deduced.

The consistency of the calibration curve was checked by measured and calculated efficiency value for ^{134}Cs and ^{133}Ba. The result is relatively good with difference about 0.6% and 0.4% for without and with absorber, respectively, and are presented in Table 2.

Typical uncertainties components for Ionization-chamber Centronic IG11/A20 are presented in Table 3.

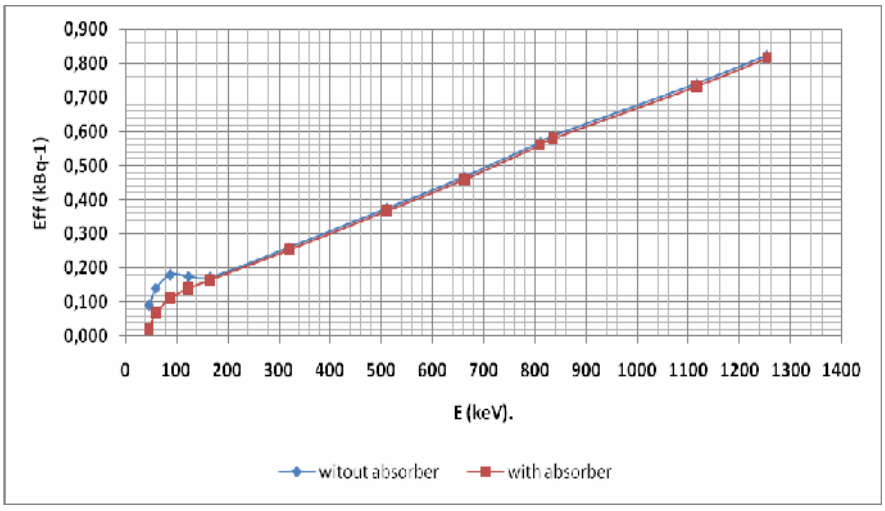

Fig. 1. Efficiency curve of the ionization chamber Centronic IG11/A20, PTKMR BATAN.

Table 2. Comparison of calculated and measured nuclide efficiency.

	Without absorber			With absorber		
	ε_N (kBq^{-1})		Δε (%)	ε_N (kBq^{-1})		Δε (%)
Nuclide	Calculated	Measured		Calculated	Measured	
^{133}Ba	0.320	0.321	-0.4	0.288	0.283	1.7
^{134}Cs	1.062	1.068	-0.6	1.052	1.046	0.6

Tabel 3. Typical uncertainty components for Ionization-chamber Centronic IG11/A20

Component of uncertainty	*u* (%)
Standard source	2.5
Current measurement	0.1
Background	0.06
Linearity of instrument	0.05
Geometry	0.5
Radionuclide impurity	0.01
Time measurement	0.01
Calibration curve extrapolation	1.5
Half live	0.1
Total uncertainty	2.96
Expanded uncertainty, *U*(*k*=2)	5.81

Conclusions

Calibrated ionization chamber measuring systems which help to establish and maintain uniformity in activity determination of photon emitting radionuclides and to improve the accuracy of activity measurements. Photon efficiency ε versus photon energy *E* with various radionuclides for re-entrant or 4πγ ionization chamber Centronic IG11/A20 were performed in PTKMR BATAN with and without Fe absorber with thickness of 1 mm. The consistency of the calibration curve was checked by measured and calculated efficiency value of without and with absorber for ^{134}Cs and ^{133}Ba. The result is relative good with difference about 0.4% to 0.6% for with and without absorber, respectively.

Acknowledgements

Authors thankful for opportunity by Research Group for Standardization of Radionuclides, Center for Technology Radiation Safety and Metrology, BATAN for their collaboration during this study so that can be done.

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The Homogeneity Test of ^{60}Co Spherical Geometry as Calibration of a Standard Source with MONITOR 4

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Abstract. **Spherical geometry standard source of ^{60}Co is intended to be used for calibration and comparisons a lot of measuring instruments in one time measurement. To be used as a standard source for this purpose, the spherical standard source of ^{60}Co is needed to be tested its homogeneity of the intensity of radiation exposure. The testing was conducted by measuring the intensity of radiation of the standard source with MONITOR 4 survey meter at eight position measurements with the same distance that considered to represent of other position. The measurement result obtained, show that standard source of ^{60}Co in the spherical geometry can be used for homogeneity of the intensity of the radiation exposure produced quite good with a difference between 0.37% up to 0.84% of average value of taking the measurement, still below the permitted limit of 1%. The difference of activity measured is 3.45% of the value activity standard source in fact, below 5% of the confidence level 95%.**

Keywords: Homogeneity, Spherical geometry, ^{60}Co.

Introduction

A survey meter is a radiation measuring instrument that is used for measuring radiation the a work place or radiation in the environment. A survey meter is used to measure the dose rate (intensity) of radiation directly in a work place. Therefore, survey meter should be portable, easy to carry in a radiation survey activity in all fields. As one of the completeness of safety instruments in the work ability of measuring survey meter should always be maintained in optimal conditions. To get the thorough and accurate measurement instrument needed to function properly, the correct method and experienced human resources. To obtain this measurement capability should survey meter be calibrated and is already a provision that any radiation measuring devices must be calibrated periodically by the user or the authorized agency. Calibration aims to determine the level of when used performance in measuring instrument and guaranteed traceability of measurement results. Calibration can determine the value associated with the performance of measuring instrument or reference material. This is achieved by direct comparison against a standard measuring instrument or certified reference materials. Output of calibration is a calibration certificate. In addition to the certificate, there are also usually labels or stickers that had been pinned on calibrated instrument. There are three important reasons why a device needs to be calibrated:

1. To ensure that the appoint of such a device in accordance with the result of other measurements.
2. To determine the accuracy of the appointment of an instrument.
3. To know the reliability of instrument, namely that the instrument can be trusted.

To be able to perform calibration services survey meter radiation measuring instrument, Standardization Laboratory of the PTKMR – BATAN make standard radionuclide of ^{60}Co in the spherical geometry Ø 1 cm by packaging plexy glass material thickness 0.5 cm with 70.000 µCi activity measurement with Capintec CRC – 7BT of ionization chamber detector on the dated January 11, 2011. The purpose of this spherical geometry that can be applied to the measurement with a lot of radiation measuring devices at once in one time measurement, in particular radiation measurements survey meter environmental radioactivity level.

To be used as a standard radionuclide source that produces an accurate measure of data, radionuclide ^{60}Co source in the spherical geometry must be tested homogeneity level radiation exposure from all directions with the same measure distances. These tests utilize radiation exposure resulting from radionuclide sources themselves are expected radiated in all directions with the same intensity (Holnisar et. al., 2009). The better the level of homogeneity of the source of radiation exposure to radionuclides in the spherical geometry is expected the better the uniformity of

response data is obtained by measuring the measuring instrument used.

On the measurement of radionuclide ^{60}Co standard in the spherical geometry that has been designed, tested and used as a measurement of the target in order to know the resulting uniformity of radiation exposure, measurements were made of eight measurement positions with the same distance, eight measurement positions is expected to represent the position of the other measurements in obtain homogeneity level radiation exposure generated by radionuclides ^{60}Co source in the spherical geometry and measurements instrument used is MONITOR 4 survey meter which can measurement millimeter scale. With this ability expected small differences in measurement data can be read.

Testing homogeneity of exposure to radiation is form of feasibility testing whether radionuclides ^{60}Co source in the spherical geometry can used as a standard source for calibration purposes and comparison survey meter radiation measuring instrument with a lot of radiation measuring devices at the point of measurement of various measuring positions.

By knowing the level of radiation exposure homogeneity prom these sources, the result of calibration and comparison performed optimally and can be justified. Radiation exposure for each measurement position is said to have a good degree of homogeneity generated when different measurements for each position does not exceed 1% of the average value of measurements (Holnisar et al., 2011).

The average value of radiation exposure homogeneity obtained of spherical geometry ^{60}Co. With Equation 1 below ^{60}Co activity can be calculated from the spherical geometry standard used.

$$A = (D.R^2)/k \quad \text{.....(1)}$$

Where : A= Activity (Ci), D= Dose rate (R/hr), K= Factor Gamma (R/hr/Ci) at a distance of 1m, for ^{60}Co is 1,327, R= Distance source to detector (m).

Materials and Methods

Radiation measuring instruments tested was MONITOR 4 survey meter. This survey meter was conditioned on optimal working conditions and set at the distance to each measurement by measuring the same distance between the source of the radionuclides ^{60}Co standard gauge, as shown in the illustration Figure 1.

Measurements were taken on eight different positions, with the number of repetition for each position of 15 times and the measurement was done at a height of 2 m above the ground so that backscattering effects of radiation could be eliminated (Shapiro, 1974).

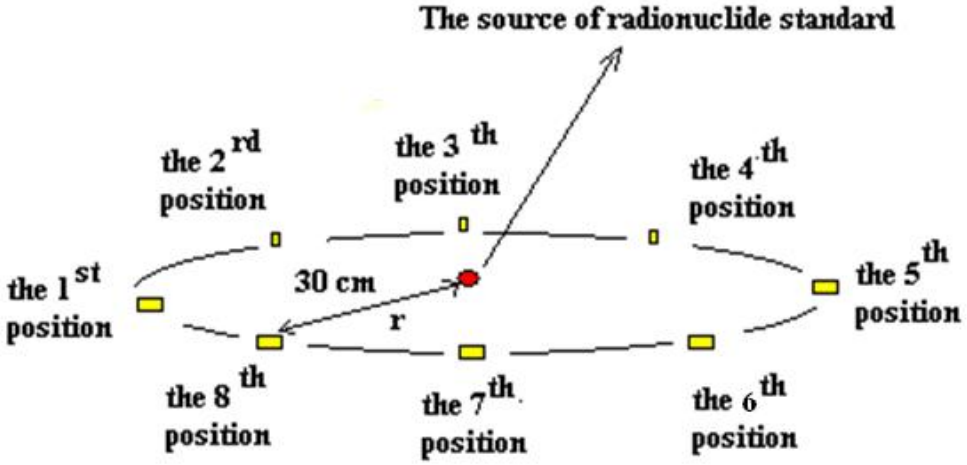

Figure 1. Illustration homogeneity measurement of radiation exposure ^{60}Co spherical geometry.

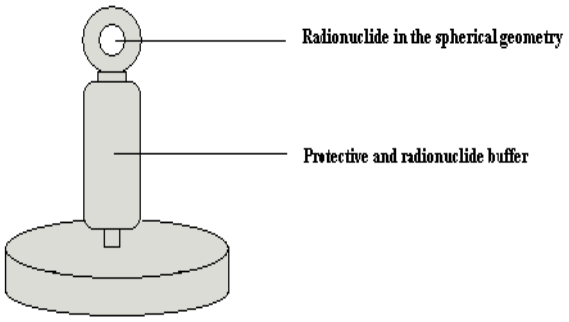

Figure 2. Radionuclide ^{60}Co spherical geometry in a protective case and a buffer plexy glass.

Results and discussion

The measurement data of homogeneity radiation exposure for the eight positions measurement by using the radiation gauge MONITOR 4 surveymeter was shown in Table 1 and 2.

Based on the measurement result have poor stability measurements because repeatability measurements performed fifteen times for each position differences obtained 20.37% above the legal limit of 5%, this can be caused from a reading of less accurate because the system still uses analog scale and the result radiation exposure measurements for each position has a different response to the data measured by the difference between 0.37% - 0.84% of the average of the measurement result. The standard radionuclide source of ^{60}Co spherical geometry can be used for calibration or comparisons with many measuring devices as well as the intensity of the radiation exposure to eight measurement positions have generated less than 1% difference. While the activity obtained from the measurements by using Equation 1 was 72.502 µCi, had a 3.45% difference in the actual activity of the standard source is 70.000 µCi.

Table 1. MONITOR 4 surveymeter measurements data from the eight position measurement.

No	mR/h							
	The 1st position	The 2rd position	The 3th position	The 4th position	The 5th position	The 6th position	The 7th position	The 8th position
1	1.00	1.00	1.10	1.10	1.00	1.20	1.10	1.00
2	1.10	1.00	1.00	1.00	1.00	1.00	1.00	1.00
3	1.10	1.10	1.10	1.00	1.20	1.00	1.00	1.20
4	1.00	1.00	1.00	1.00	1.10	1.10	1.00	1.00
5	1.10	1.10	1.10	1.00	1.00	1.10	1.10	1.10
6	1.00	1.20	1.30	1.20	1.00	1.00	1.20	1.10
7	1.10	1.10	1.10	0.90	1.10	1.00	1.20	1.00
8	1.20	1,.00	1.00	1.00	1.10	1.10	1.00	1.10
9	1.00	1.00	1.00	1.10	1.00	1.10	1.00	1.10
10	1.10	1.00	1.10	1.10	1.10	1.20	1.10	1.10
11	1.00	1.10	1.00	1.00	1.00	1.00	1.00	1.10
12	1.10	1.20	1.10	1.20	1.10	1.00	1.10	1.10
13	1.00	1.10	1.10	1.10	1.30	1.10	1.10	1.00
14	1.10	1.10	1.20	1.20	1.10	1.10	1.20	1.00
15	1.00	1.00	1.00	1.10	1.00	1.10	1.10	1.10
$\overline{X}$	1.060±0.061	1.067±0.070	1.080±0.083	1.067±0.087	1.073±0.085	1.073±0068	1.080±0.075	1.067±0.60

Table 2. The mean results of measurements at eight positions.

No	Position measurement	The measurement results (mR/h)
1	Position 1	1.060
2	Position 2	1.067
3	Position 3	1.080
4	Position 4	1.067
5	Position 5	1.073
6	Position 6	1.073
7	Position 7	1.080
8	Position 8	1.067
$\overline{X}$		1.071 ± 0.007

With the ability to measure survey meter always maintained the expected effort for the radiological safety of radiation workers or the environment done well as exposure to radiation in the environment is always monitored radiation workers with the ability to measure accurate survey meter. So efforts to provide radiological protection to workers receiving radiation to radiation doses can be reduced as low as possible (Knoll, 1999).

Conclusions

Testing the homogeneity measurement of the source of radiation exposure to radionuclides ^{60}Co standard spherical geometry, this source can be used as a standard source for calibration purpose and comparisons with many measurement instruments as well as exposure gained from testing for eight positions of the source has a difference 0.37% – 0.84% to average value measurement that was below 1%, showing a standard ^{60}Co radionuclide source in the spherical geometry had a homogeneity of radiation exposure is quite good and have different activity measured 3.45% of the value of the actual standard, under 5% of the 95% confidence level.

With the successful testing of the homogeneity of the ^{60}Co spherical geometry, Standardization Laboratory PTKMR – BATAN can perform calibration services or comparisons survey meter for environmental radioactivity level.

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DISCUSSION

Mapping of Environmental Gamma Radiation Dose Rate in West Sumatera Province

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Abstract. **Mapping of environmental gamma radiation in the province of West Sumatera has been done. This activity is part of mapping the environmental radiation in Indonesia in order to complet the baseline of Indonesia's environmental radiation data. Measurements were carried out by using a portable gamma spectrometer Exploranium GR-130, and sampling area was divided into the 40 km x 40 km areas using GPS positioning. Environmental gamma radiation dose rate in the region of West Sumatera was varied from 35.00±10.48 nSv/h to 103±23 nSv/h with an average of 60±13 nSv/h. Environmental gamma radiation levels in the region of West Sumatera were relatively equal to the average level of gamma radiation in Indonesia, but it was slightly higher compared to the gamma radiation measurements obtained in most areas of Java, most of Sumatera, Kalimantan, Bali and Nusa Tenggara islands.**

Keywords: dose rate, gamma radiation, environmental, West Sumatera

Introduction

According to the UNSCEAR report of 2000, approximately 84% of the total annual radiation dose received by the population of the world comes from natural sources of radiation coming from the earth's crust or also called radionuclides primordial (UNSCEAR, 2000).

Primordial natural radionuclides are the type of radioactive nature that has been formed since the formation of the earth, or often abbreviated as NORM (Naturally Occurring Radioactive Materials). These radioactive elements includes U-238, Th-232 along with its exuviate (also known as U-238 series and Th-232 series) and K-40. Natural radioactive concentration varies depending on the type of soil and rock or its geologial formation (Henriksen and Maillie, 2003). Natural radioactive concentration will increase if a mining activity exist, where the radioactive elements will be mobilized from the earth's crust during the process or commonly known as TENORM (Technologically Enhanced Naturally occurring Radioactive Materials) (Henriksen and Maillie, 2003; IAEA, 2004).

Given Indonesia is rich in mineral deposits and an increase in utilizing minerals in various industries, it is necessary to mapping natural radiation levels as a baseline data. Center for Technology Radiation Safety and Metrology (PTKMR) in its 2010-2014 Strategic Plan has a program in mapping environmental radioactivity level in Indonesia, one of this mapping of radioactivity is in the Province of West Sumatra.

West Sumatra is a province on the island of Sumatra, which lies between 0.54 and 3.30 LU and LS 980 36'BT and 1010 53 'E, with an area of 42,012.89 km^2. In general, West Sumatra has geological formations of alluvium, andesite, basalt, batholiths, limestone and volcanic rocks, as well as a small fraction of granite. West Sumatra has the potential of minerals such as manganese, gold, iron ore, coal, granite and quarry materials lainnya (Peta Digital, 1995).

Materials and Methods

Determination of measurement location

Determination of the location of the dose rate measurements was performed by dividing the area of West Sumatra into the box or cell of 40 km which is also called grid of 40 km x 40 km. Sampling points were done by systemic random sampling methods. This systemic method of random sampling have been selected to facilitate the implementation of direct measurements in the field with Global Positioning System (GPS) models made by Garmin GPSMAP 60CSx-USA (IAEA, 2004; Bowen, 1979; Garmin, 2001), where the location of the measurement is not fixated on one point that is determined but depends on the location of sampling that can be reached but still within the measurement box (40 x 40 km grid). Image measurement locations of Western Sumatra and the method of determining the location is shown in Figure 1.

Measurement of gamma dose rate

Measurement of environmental gamma dose rate radiation performed using the portable gamma spectrometer Exploranium Radiation Detection System Model GR-130-mini-SPEC with detector NaI (Tl) which can be seen in Figure 2. This instrument is equipped with several electronic devices support, the high voltage (HV), ratemeter, automatic timer with preset time, the power of the battery, LCD displays,

and a joystick that functions to operate. The measuring instrument can be set (setting) every second for 2 to 3 hours (Exploranium, 2001).

Dose rate measurements were performed at each location and has been determined and geographical position was stored using GPS (Global Positioning System), as shown in Figure 2. Measurement of gamma dose rate of radiation-use environment Exploranium GR130 is very practical to take to the field because it had a fairly small size, ie length 235 mm, width 110 mm, and 170 mm heigh (including the handle) and has a weight of 1.9 kg (without battery) or 2.4 kg (including 2 batteries). Units shown by the instruments is nSv/hour.

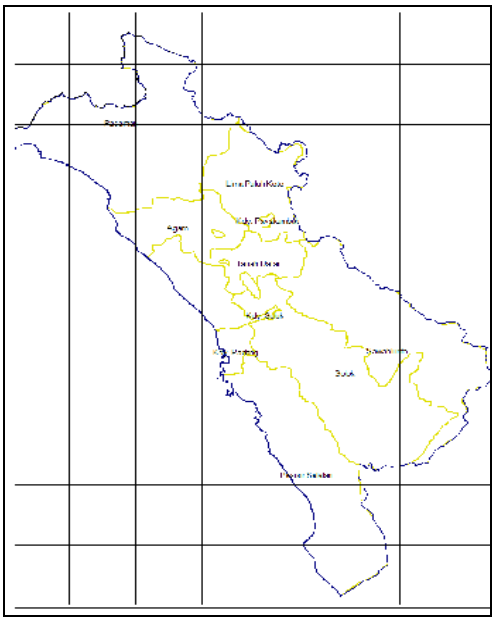

Figure 1. Map of West Sumatra for data collection that consists of 40 km x 40 km grids.

Figure 2. Surveymeter Exploranium GR130 portable (left) and GPS (*Global Positioning System*) Made by Garmin Model GPSMap 60CSx (right).

Results and Discussion

Result of gamma radiation dose rate measurement in 28 locations in West Sumatra is presented in Table 1 and map of environmental gamma radiation dose rate (represenetd as nano Sievert per hour, nSv/h) meaurement of West Sumatra Province is shown in Figure 3.

Table 1 and Figure 3 shows that the amount of gamma radiation dose rate varies with the range between 35 ± 10 nSv/hour to 103 ± 23 nSv/h with the mean of 60 ± 13 nSv/hour. The lowest dose rate of gamma radiation of 35 ± 10 nSv/h was measured at Kec. Kinali, West Pasaman with geological formations of andesite from Talamau Mount. Whereas the highest gamma radiation dose rate of 103 ± 23 nSv/h was obtained in Palupuh district, Agam Regency with geological formations of rock Amas volcano with volcanic deposits, subaerial (IAEA, 2004).

Table 2 shows that the level of gamma radiation in the province of West Sumatra relatively high nationally. Environmental gamma radiation level in West Sumatra with relatively lower levels of gamma radiation in the 5 regions, namely Central Java, North Sumatra, South Sulawesi, West Sulawesi and Ambon island. However, for all Sumatera areas, environmental gamma radiation level in West Sumatra is relatively higher when compared to other regions. The highest gamma radiation levels measured in Sumatra Island are North Sumatra (74 ± 37 nSv/h) and West Sumatra and then followed by other lower areas.

Conclusions

Results of measuring the level of environmental gamma radiation dose rate at 28 locations in West Sumatra varies from 35 ± 10 nSv/h to 103 ± 23 nSv/h with an average of 60 ± 13 nSv/h. Environmental gamma radiation level in the region of West Sumatra are relatively equal to the average level of gamma radiation in Indonesia. The mapping results also indicate that the level of gamma radiation in West Sumatra is slightly higher than the results of measurements of gamma radiation in most areas of Java, most of Sumatra, Kalimantan, Bali and Nusa Tenggara.

Acknowledgements

We thank to all members of teamwork that already contributed to this research and done well.

Table 1. Results of gamma radiation dose rate measurement.

Code	Position		Dose Rate (nSv/h)	Locations
	South Lattitude	East Logitude		
TSB-1	-0.55140	100.31174	99 ± 17	Kec. Lingkung Rayotanam - Kab. Pariaman
TSB-2	-0.81227	100.30756	62 ± 19	Kec. Koto Tangah - Kab. Pariaman
TSB-3	-1.25790	100.52466	44 ± 11	Kec. Bayang - Kab. Pesisir Selatan
TSB-4	-1.61360	100.64561	50 ± 11	Kec. Sutera - Kab. Pesisir Selatan
TSB-5	-1,.67427	100.72835	65 ± 14	Kec. Lengayang - Kab. Pesisir Selatan
TSB-6	-2.01521	100.95961	61 ± 14	Kec. Indrapura - Kab. Pesisir Selatan
TSB-7	-2.24729	101.12762	62 ± 14	Kec. Lunang Silau - Kab. Pesisir Selatan
TSB-8	-2.09766	101.24119	59 ± 13	Kec. Muaro Sako - Kab. Pesisir Selatan
TSB-9	-1.52184	101.27080	35 ± 10	Kec. Sangir - Kab. Solok Selatan
TSB-10	-1.53802	101.56827	59 ± 13	Kec. Talang air - Kab. Solok Selatan
TSB-11	-1.09720	101.84710	43 ± 12	Kec. Koto Baru - Kab. Dharmasraya
TSB-12	-1.16793	101.62717	59 ± 13	Kec. Sungai Rumbai - Kab. Dharmasraya
TSB-13	-1.03178	101.61836	51 ± 15	Kec. Sungai Dareh - Kab. Dharmasraya
TSB-14	-0.86615	101.29588	78 ± 14	Kec. Kamang Baru - Kab. Sawah Lunto
TSB-15	-0.14933	100.28139	103 ± 23	Kec. Palupuh - Kab. Agam
TSB-16	0.10666	100.17902	86 ± 13	Kec. Lubuk Sikaping - Kab. Pasaman
TSB-17	0.54832	100.06850	56 ± 11	Kec. Rao - Kab. Pasaman
TSB-18	0.60741	99.93312	37 ± 9	Kec. Rao - Kab. Pasaman
TSB-19	0.14243	99.92147	36 ± 9	Kec. Sei Abuah - Kab. Pasaman Barat
TSB-20	-0.05218	99.90015	35 ± 10	Kec. Kinali - Kab. Pasaman Barat
TSB-21	-0.32122	99.96596	88 ± 14	Kec. Lubuk Basung - Kab. Agam
TSB-22	0.06128	100.75266	70 ± 12	Kec. Labuh Silang - Kab. Payakumbuh
TSB-23	-0.47825	100.60686	68 ± 11	Kec. Lima Raum - Kab. Tanah datar
TSB-24	-0.63135	100.88685	40 ± 10	Kec. Koto VII - Kab. Sawahlunto
TSB-25	-0.67960	101.09428	71 ± 12	Kec. Sijunjung - Kab. Sawahlunto
TSB-26	-0.81588	100.78448	39 ± 9	Kec. Sungai durian - Kab. Solok
TSB-27	-0.79186	100.63461	56 ± 12	Kec. Tanah Garam - Kab. Solok
TSB-28	-1.23589	100.87467	57 ± 14	Kec. Surian - Kab. Solok

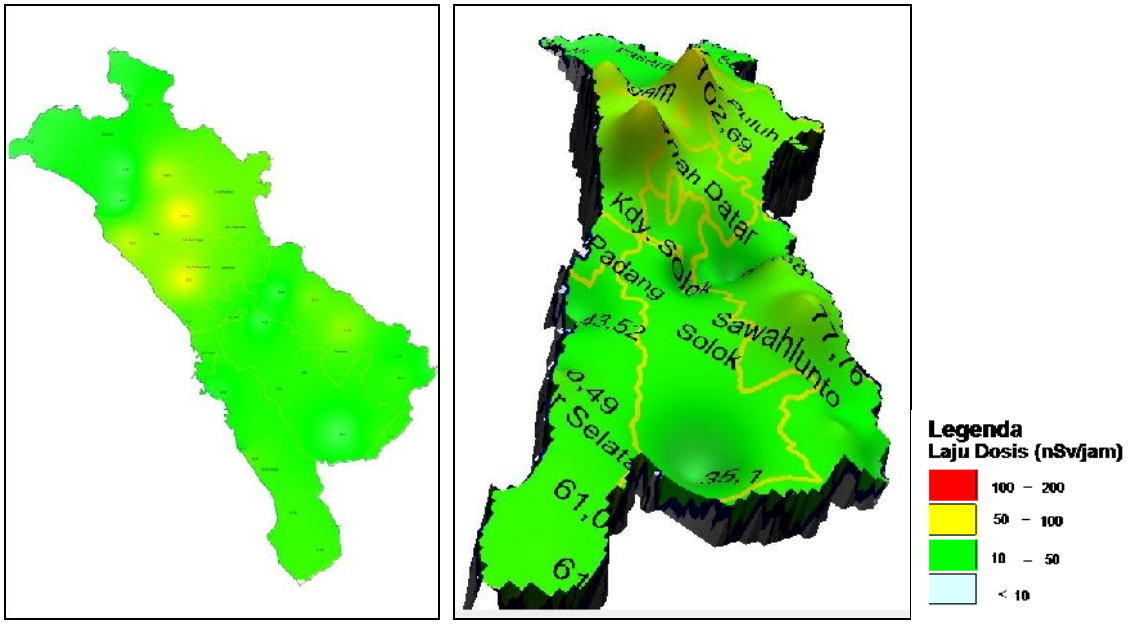

Figure 3. Map of gamma radiation dose rate in West Sumatra Province.

Table 2. Data environmental gamma radiation dose rate averages in some areas in Indonesia.

No.	Measurement area	Dose Rate (nSv/h)	Annotation
1.	Lampung, Sumatera	48 ± 11	Maksun et al., 2006
2.	Bengkulu, Sumatera	43 ± 10	Maksun et al., 2006
3.	North Sumatra	74 ± 37	Maksun et al., 2006
4.	Riau, Sumatera	44 ± 20	Maksun et al., 2006
5.	Jambi, Sumatera	27 ± 4	Maksun et al., 2006
6.	South Sumatra	45 ± 12	Maksun et al., 2006
7.	West Java	39 ± 10	Suharyono et al., 2007
8.	Central Java	82 ± 46	Suharyono et al., 2007
9.	East Java	31 ± 8	Suharyono et al., 2007
10.	Yogyakarta	39 ± 8	Suharyono et al., 2007
11.	Banten	40 ± 7	Suharyono et al., 2007
12.	DKI Jakarta	50 ± 1	Suharyono et al., 2007
13.	South Kalimantan	51 ± 19	Sutarman et al., 2008
14.	West Kalimantan	62 ± 17	Sutarman et al., 2008
15.	East Kalimantan	46 ± 14	Sutarman et al., 2008
16.	Central Kalimantan	59 ± 15	Sutarman et al., 2007
17.	South Sulawesi	118 ± 1	Sutarman et al., 2007
18.	Southeast Sulawesi	61 ± 1	Sutarman et al., 2007
19.	West Sulawesi	272 ± 3	Sutarman et al., 2007
20.	Bali island	29 ± 2	Sutarman et al., 2009
21.	Nusa Tenggara Barat	43 ± 1	Sutarman et al., 2009
22.	Nusa Tenggara Barat	30 ± 2	Sutarman et al., 2009
23.	Ambon Island	72 ± 23	Kusdiana et al., 2010

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The Quantitative Determination of Tumor Marker Cancer Antigen 15-3 in Serum of Pre and Post Chemoradiotheraphy of Metastatic Breast Cancer Patients by ELISA and IRMA Methods

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Abstract. **Serum tumor markers are molecules or substances that are released by a tumor into the blood circulation that can be detected and quantitated. Cancer Antigen 15-3 (CA 15-3) is a mucinous carbohydrate antigen product of the MUC1 gene which known as a serum tumor marker for breast cancer. The quantitative determination of CA 15-3 serum by immunoassay are well-characterized assay that allow to used for the detection of circulating CA 15-3 in peripheral blood. Aim of this study was to prospectively evaluate whether a radiochemotherapy given to the metastatic breast cancer patients give positive response by serial quantitative determination of CA 15-3 in serum. Sera of 33 women with metastatic breast carcinoma obtained pre and one month post chemoradiotherapy were assayed for CA 15-3 by ELISA and IRMA methods. We found that the mean serum CA 15-3 levels in patients before therapy were significantly higher (50.4758 U/mL by ELISA, normal value : ≤ 35 U/mL and 59.0 U/mL by IRMA, Normal value : ≤ 30 U/mL) compared with those of CA 15-3 after therapy (44.5697 U/mL by ELISA and 46.067 U/mL by IRMA). Although serum levels of CA 15-3 assayed by both method, before and after treatment were decreasing significantly ($p < 0.000$ by ELISA and $p < 0.001$ by IRMA) but those were still beyond of cut off normal value. Compared to ELISA and IRMA methods, the assay results of CA 15-3 serum were statistically significant different for both sample groups ($p < 0.000$ for both pre and post teraphy sample groups). We concluded that decreasing concentrations of CA 15-3 one month after treatment compared with before treatment maybe used as the first starting of predictive tool for monitoring of cancer progression. And for evaluating the response of treatment (partial response, total response, stable disease, *de novo* disease progression (PD) or secondary PD, we need to determine CA 15-3 serum levels monthly after treatment for up to 6 months. In comparison of CA 15-3 serum assay by ELISA and IRMA methods gave significantly different result.**

Keywords : Tumour marker CA 15-3 serum, pre and post chemoradiotheraphy of breast cancer, ELISA and IRMA method

Introduction

In Indonesia, the exact incidence and prevalence of cancer are still not known because population based of cancer registries have not done yet. The latest fact is that cancer has risen to become the fifth position in rank of the disease causes death after infectious diseases, cardiovascular diseases, traffic accidents, nutritional deficiency and congenital diseases. Data collected from various hospitals in big cities in Indonesia shows that cancer incidence increased by 2–8% per year (Tjindarbumi, 2002; Aziz, 2009). Data which have been collected from 13 pathological laboratories throughout Indonesia during the period of 1988–91 show that in the combined picture (male and female), cervical, breast, skin, rectum and nasopharynx are the five major anatomical sites for cancer disease (Tjindarbumi, 2002). Histopathological report in 2002 revealed that cervical cancer, breast cancer and ovarian cancer were the most frequent cancer among female 2,532 cases, 2,254 cases and 829 cases, respectively. Indonesia has categorized as a low middle-income developing country, especially after the economic crises hit Indonesia in 1998 (Aziz, 2009).

The breast is mainly composed of fatty tissue called adipose tissue. Within this tissue is a network of lobes, which are made up of tiny, tube-like structures called lobules that contain milk glands. Tiny ducts connect the glands, lobules, and lobes, carrying the milk from the lobes to the nipple, located in the middle of the areola (darker area that surrounds the nipple of the breast). Blood and lymph vessels run throughout the breast; blood nourishes the cells, and the lymph system drains bodily waste products. The lymph vessels connect to lymph nodes, which are tiny, bean-shaped organs that normally help fight infection (Figure 1) (American Cancer Society, 2013).

Actually, breast cancer can be totally cured if it is caught early enough by methods of detecting of breast cancer when it's still the size of the head of a pin. But symptoms of breast cancer often don't occur until the late stages, when it has already spread to other glands and organs. Known that stage at diagnosis has very close relation with 5-year relative

survival, where localized stage (confined to primary site) has 5-year relative survival 98.6% whereas regional advance (spread to regional lymph nodes) and distant stage (cancer has metastasized) has 5-year relative survival 84.4% and 24.3% respectively (National Institute of Health and National Cancer Institute, 2013).

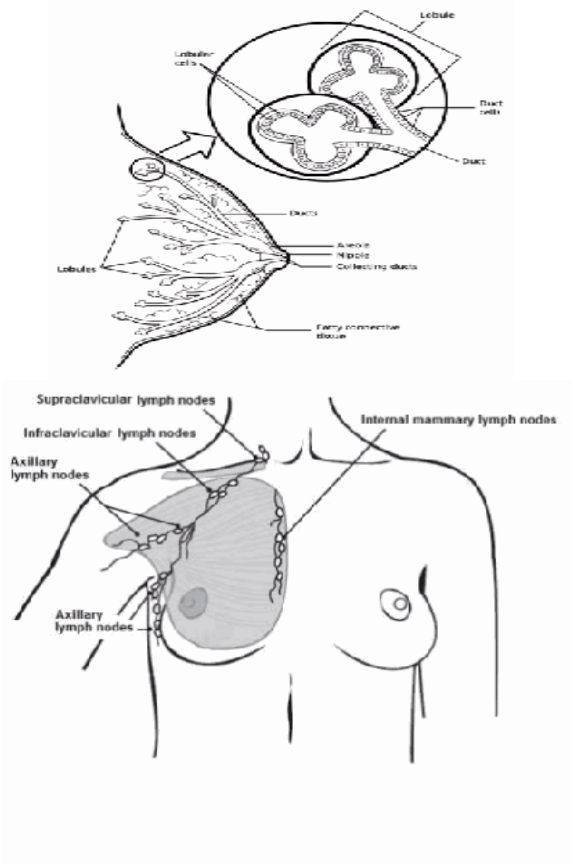

Figure 1. The breast anatomy

Several tumour markers in peripheral blood for the detection of breast cancer cells have been described by investigators in many result studies. Commonly assessed mRNA markers include CK18, CK19, CK20, Mucin-1 (MUC-1), and carcinoembryonic antigen (Jacobs, 1989). However, recent studies have shown several of these markers to be expressed in normal cells of peripheral blood, lymph nodes, and/or bone marrow yielding false-positive results. Furthermore, many of these tumour / molecular markers are also expressed frequently in normal epithelial cells. In addition, no one tumor marker is consistently and specifically expressed by all of the primary tumors for a particular malignancy, and marker expression may vary between a primary tumor and its metastasis. These findings may contribute to the lack of consistent correlations between any one tumor marker and well-known clinical and pathological prognostic factors.

There are a number of tumor markers that can help clinicians to identify and diagnose which breast cancer patients will have aggressive disease and which will have an indolent course. These markers include estrogen and progesterone receptors, DNA ploidy and percent-S phase profile, epidermal growth factor receptor, HER-2/neu oncogene, p53 tumor suppressor gene, cathepsin D, proliferation markers and CA15-3 (Abcam, 2012).

CA 15-3 is a mucinous carbohydrate antigen product of the MUC1 gene, originally identified by two monoclonal antibodies: DF3 raised against a membrane fraction of breast liver metastases and 115D8 raised against milk fat globule membrane. The antigenic component is an epitope of Polymorphic Epithelial Mucin (PEM), a high molecular weight (300-400 kDa) heavily glycosylated protein encoded by the MUC1 gene. The polymorphism is associated with protein components that are expressed in repeated sequences that can vary between 20 and 75 times. Recent data suggest that MUC1 plays a role in cell adhesion (leading to decreased cell-cell and cell-extracellular matrix interactions).

CA15-3 is most useful for monitoring patients post-operatively for recurrence, particularly metastatic diseases. 96% of patients with local and systemic recurrence have elevated CA15-3, which can be used to predict recurrence earlier than radiological and clinical criteria (Hollingsworth, 1994). A 25% increase in the serum CA15-3 is associated with progression of carcinoma. A 50% decrease in serum CA15-3 is associated with response to treatment. CA15-3 is more sensitive than CEA in early detection of breast cancer recurrence. In combination with CA125, CA15-3 has been shown to be useful in early detection of relapse of ovarian cancer (Bast, 2003; Rustin, 1992). CA15-3 levels are also increased in colon, lung and hepatic tumors. Moreover, not much is known regarding the presentation of breast cancer of Asian women particularly in developing South East Asian nations. The objective of this study is therefore to compare the clinical and pathological characteristics of breast cancer patients between two tertiary public hospitals

Materials and Methods

Patients and tumor markers

Patients are newly diagnosed breast cancer patients who will receive chemoradioteraphy treatment at Fatmawati Hospital during year 2012. Tumor marker Cancer Antigen 15-3 level in serum of pre and one month post chemoradiotheraphy breast cancer patients were determined by non isotopic immunoassay (Enzymme Linked Immuno Sorbent Assay / ELISA) and isotopic immuno assay (Immuno Radio Metric Assay / IRMA) methods.

Procedure of the ELISA sandwich method.

The CA15-3 ELISA test is based on the principle of a solid phase enzyme-linked immunosorbent assay. The assay system utilizes a monoclonal antibody, directed against a distinct antigenic determinant on the intact CA15-3 molecule, for the solid phase immobilization (on the microtiter wells). A rabbit anti-CA15-3 antibody conjugated to horseradish peroxidase (HRP) is in the antibody-enzyme conjugate solution. The test sample is allowed to react sequentially with the two antibodies, resulting in the CA15-3 molecules being sandwiched between the solid phase and enzyme-linked antibodies. After two separate 1-hour incubation steps at 37°C, the wells are washed with wash buffer to remove unbound labeled antibodies. A solution of TMB Reagent is added and incubated for 20 minutes, resulting in the development of a blue color. The color development is stopped with the addition of Stop Solution and the color is changed to yellow. The concentration of CA15-3 is directly proportional to the color intensity of the test sample. Absorbance is measured spectrophotometrically at 450 nm. Step of reactions of ELISA is presented in Figure 2.

Calculation of results was done manually according to standard procedure.

Procedure of the IRMA method

The principle of the assay is a one step non-competitive sandwich immunoradiometric assay method. The present method employs several monoclonal anti-CA 15-3 antibodies which recognize two different epitopes of the molecules. One antibody is absorbed in solid phase (coated tube), the others (labeled with Iodine^{125}, labeled with biotin) are used as tracer and tracer buffer. The sample to be tested, is incubated in the coated tube, following the incubation, after aspiration and washing, the labeled antibody is added to the coated tubes, where it binds to the solid phase, by means of the antigen in standards and samples. The amount of bound tracer will thus be directly proportional to the antigen concentration. After a further aspiration and washing cycle, the residual radioactivity in the tubes is measured by a gamma counter. Note that for re-assaying samples with concentrations greater than 200 U/mL, dilute the sample an additional 1 : 5 or 1 : 10 using the serum diluent. Multiply by the dilution factor to obtain the concentration of the samples. Calculation of results was done manually according to standard procedure.

Statistical analysis

Descriptive and inferential statistic analysis have been done on two sets paired data for hypothesis evaluation by SPSS ver. 16 software.

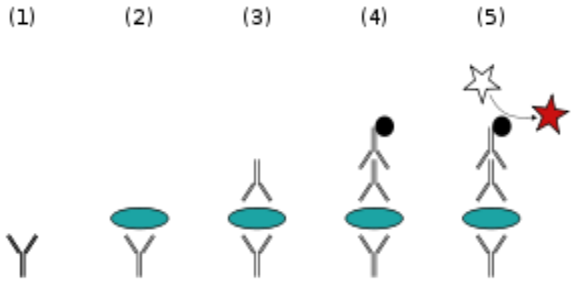

Figure 2. Steps of reaction of ELISA.

Results and Discussion

Results of standards ELISA assay run with optical density (OD) reading at 450 nm shown in the Y-axis against CA15-3 concentrations shown in the X-axis are presented in Table 1 and Figure 3.

Results of standards IRMA assay run with B/Bmax (%) shown in the Y-axis against CA15-3 concentrations shown in Table 2 and Figure 4.

For Hook effect in IRMA sandwich assay, if samples with very high concentration of CA 15-3 (with this kit is if more than 5000 U/mL) are assayed undiluted, the hook effect can cause apparent values of concentration lower than the real ones.

Table 1. Standard curve of CA 15-3 by ELISA method.

Conc. of CA15-3 (U/ml) X-axis	**Absorbance @450 nm (OD)** Y-axis
0	0.021
15	0.425
30	0.693
60	1.214
120	1.956
240	2.845

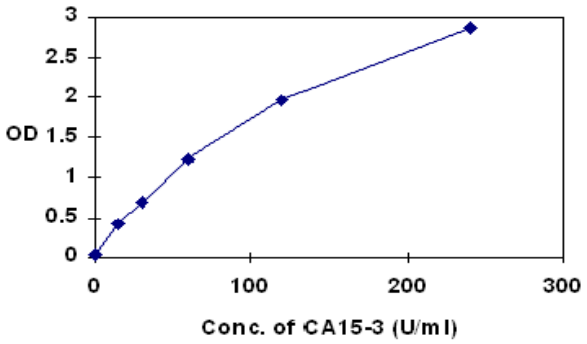

Figure 3. Graph of the CA 15-3 by ELISA method.

Table 2. Standard curve of CA 15-3 by IRMA method.

Conc. of CA15-3 (U/ml) X-axis	Cpm	B/Bmax (%) Y-axis
Total activity	257144	
0	2119	7.15
25	8447	21.74
50	11433	38.55
100	18991	84.04
200	29855	100

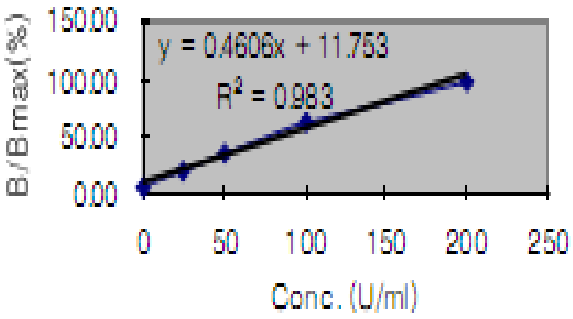

Figure 4. Graph of the CA 15-3 by IRMA method.

Table 3. Concentration of CA 15-3 in serum level in blood measured by ELISA and IRMA methods.

No.	Age (years)	ELISA method (A)		IRMA method (B)	
		Conc of CA15.3 Pre-therapy U/mL	Conc of CA15.3 Post-therapy U/mL	Conc of CA15.3 Pre-therapy U/mL	Conc of CA15.3 Post-therapy U/mL
1.	33	243	206	270.5	189
2.	63	37	33	43.1	30.1
3.	64	30	27	35.5	24.9
4.	35	11.6	10.5	13.7	9.7
5.	49	22.1	19.9	26	18.2
6.	75	9.1	8.1	10.1	7.8
7.	56	41.2	37	48.5	37.3
8.	46	11	10.4	13.6	9.6
9.	55	27.5	24.8	32.4	22.8
10	34	55.9	50	65.8	46.1
11	41	60.2	42	70.9	49.6
12	50	128	115.9	151.5	106.5
13	41	194.3	174	228.6	160
14	28	127.8	113	150.3	115.6
15	58	17.3	15.5	20.3	14.2
16	39	43.1	38.7	50.6	38.9
17	54	20.1	18.1	23.7	18.2
18	53	15.6	13	19.6	15.1
19	31	82.9	74.5	97.5	68.3
20	48	121	108.9	143.3	110.2
21	36	76.5	69	**90**	**117**
22	46	10.5	9.5	12.4	16.1
23	55	21	18.9	24.7	19.0
24	33	44.7	40	52.6	40.5
25	61	49.3	44.3	**58.0**	**75.4**
26	48	17.8	16	**20.9**	**27.2**
27	51	22	20.1	25.5	19.6
28	42	43	38.7	50.6	38.9
29	76	8.7	7.9	10.3	7.9
30	45	5.5	5	6.5	5
31	48	11.1	9.9	13	10
32	43	18.2	16.4	21.5	16.5
33	52	38.7	34.8	45.5	35
Mean ± SE		**50.48 ± 9.61**	**44.57 ±8.39**	**59.00 ±11.03**	**46.07 ± 8.12**

Tumour marker of CA 15-3 serum level has been analyzed in 33 patients before and after receiving treatment chemoradiotherapy at Fatmawati Hospital in Jakarta during year 2012. The median age of the patients was 48 years (range 28-76 years), and the results are presented in Table 3.

From Table 3 and Figures 5 and 6 it can be seen that serum CA-15-3 levels are higher for pre-therapy compared to post therapy measured both by ELISA or IRMA except for 3 cases (case 21, case 25 and case 26) detected with IRMA method. The mean serum CA 15-3 levels in patients before therapy were significantly higher (50.4758 U/mL by ELISA, Normal value : ≤ 35 U/mL and 59.0 U/mL and by IRMA, Normal value : ≤ 30 U/mL) compared with those of CA 15-3 after teraphy (44.5697 U/mL by ELISA and 46.067 U/mL by IRMA). Although serum levels of CA 15-3 assayed by both method, before and after treatment, were decreasing significantly ($p < 0.000$ by ELISA and $p < 0.001$ by IRMA) but these were still beyond of cut off normal value. Comparison of ELISA and IRMA method give the assay results of CA 15-3 serum statistically significant different for both sample groups ($p < 0.000$ for both pre and post therapy sample groups).

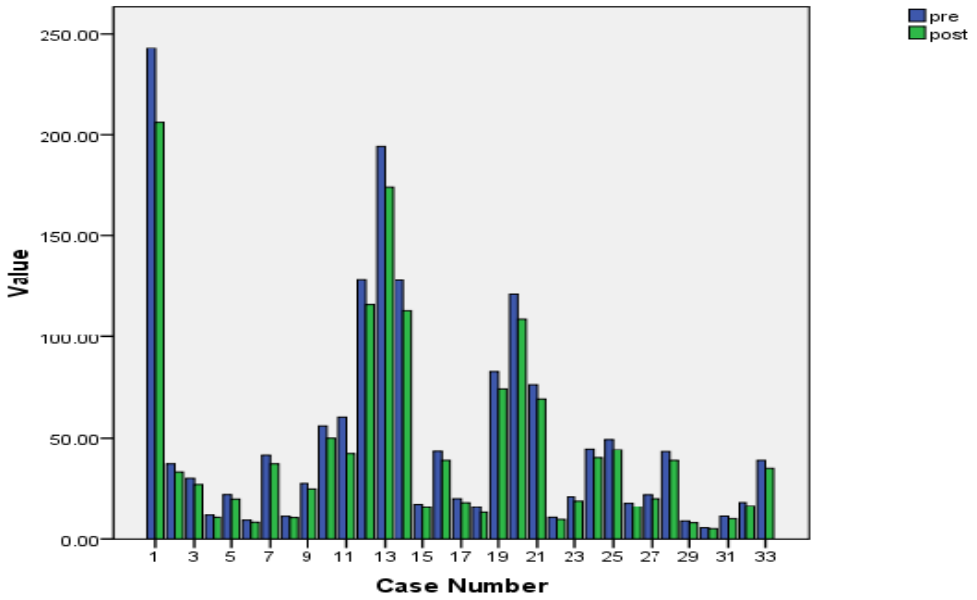

Figure 5. Bar chart of CA 15-3 assay results by ELISA method.

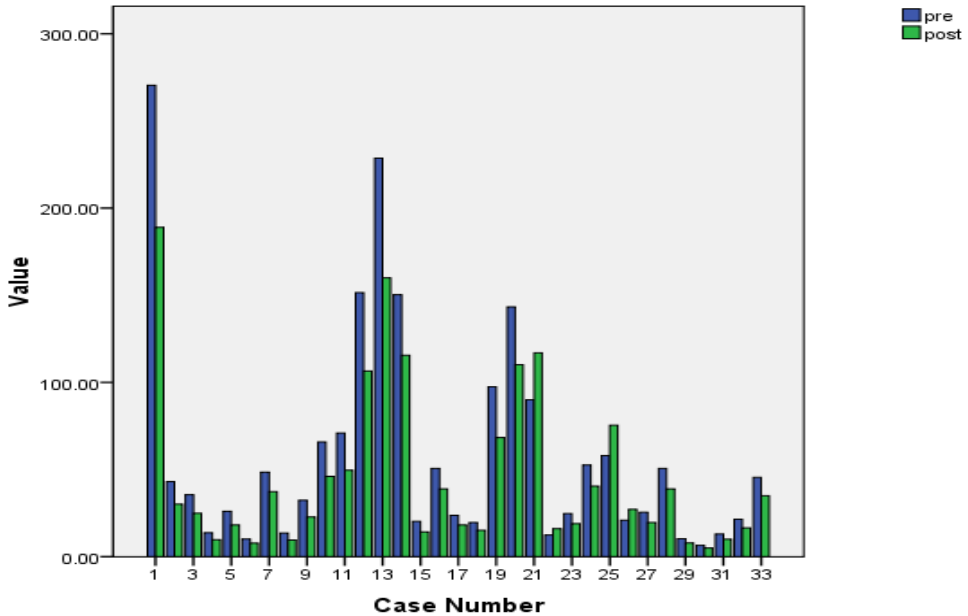

Figure 6. Bar chart of CA 15-3 assay results by IRMA method.

Table 4. Statistic analysis of CA 15-3 assay results by ELISA and IRMA method (Paired Samples Correlations).

	Test	N	Correlation	Significant
Pair 1	Pre by ELISA & pre by IRMA	33	.999	.000
Pair 2	Post by ELISA & post by IRMA	33	.972	.000

Tumor markers are substances that can be found in the body when cancer is present. The classic tumor marker is a protein that can be found in the blood in higher than normal amounts when a certain type of cancer is present, but not all tumor markers are like that. Some are found in urine or other body fluid, and others are found in tumors and other tissue. They may be made by the cancer cells themselves, or by the body in response to cancer or other conditions. Most tumor markers are proteins, but some newer markers are genes or other substances. Tumor markers alone are rarely enough to show that cancer is present. Most tumor markers can be made by normal cells as well as by cancer cells. Sometimes, non-cancerous diseases can also cause levels of certain tumor markers to be higher than normal. And not every person with cancer may have higher levels of a tumor marker (De Laine, 2008).

Tumour markers are not recommended for screening asymptomatic patients for malignancy because they generally lack specificity where many patients may have an elevated result due to benign disease and lack sensitivity where many patients with malignancy will have a normal result. An inappropriately ordered test that returns an elevated result, can lead to a cascade of unnecessary investigations, whereas a negative result may give false reassurance. There is evidence that tumour markers are not always requested appropriately. A 12 month study of tumour marker requesting, found that in the majority of instances, tumour markers were being inappropriately requested as screening or diagnostic tests. In addition, approximately 20% of all requests for CA 125 and CA 15-3 (both usually indicated only in women) were requested in men (McGinley, 2003).

Conclusions

1. Decreasing concentrations of CA 15-3 one month after treatment compared with before treatment maybe can used as the first starting of predictive tool for monitoring of cancer progression.
2. For evaluating the response of treatment (partial response, total response, stable disease, *de novo* disease progression (PD) or secondary PD, we need to determine CA 15-3 serum levels monthly after treatment for up to 6 months. Based on CA 15-3 assay, ELISA and IRMA methods give significantly different result.

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Review of Regulatory Control of Waste Management of Ionization Chamber Smoke Detector in Indonesia

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Abstract. **Review of regulatory control of waste management of ionization chamber smoke detector (ICSD) was carried in the framework of the restructuring of Chairman Regulation (CR) of Nuclear Energy Regulatory Agency (BAPETEN) arranging the technical requirements in detail. CR of BAPETEN draft has been compiled since 2011 and radioactive waste controlling approach can provide radiation safety assurance to the member of the public and environment. Radioactive substance used is americium (Am-241), the activity ≤ 40 kBq, but also up to ≤ 2.2 GBq, value is far above exemption level (10 kBq). Even, the first ICSD made, using some radionuclides, among of them is radium (Ra-226) and it is more dangerous than the Am-241. In addition, radiation sign mounted on the inside of the ICSD can cause public consternation if disused ICSD disposed indiscriminately. Study was conducted through literature, for example in European Union, depond on the member states, disused ICSD are managed as hazardous waste or radioactive waste but with the provision that radionuclide is Am-241 and maximal activity is 40 kBq. For while in Australia, disused ICSD must be managed as radioactive waste if the number more than 10 products. Conclusions of the study are radioactive substances that can be used only Am-241, maximal activity is 40 kBq and number of them are not necessary to be limited, for further disused ICSD must be collected by the distributor or user for further managed on Radioactive Waste Management Centre-National Atomic Energy Agency (PTLR-BATAN) in Serpong, Banten Province.**

Keywords: ICSD, consumer products, radioactive waste, radiation safety.

Introduction

Ionization chamber smoke detector (ICSD) is the one of consumer products which containt radioactive. According to the International Atomic Energy Agency (IAEA) in new Basic Safety Standards (BSS) GSR Part 3, the definition of consumer productsis a device or manufactured item into which radionuclides have deliberately been incorporated or produced by activation, orwhich generates ionizing radiation, and which can be sold or made available to members of the public without special surveillance or regulatory control after sale. From point of view of the IAEA's definition can be interpreted that the use of consumer products can be exempted from regulatory cotrol but transport and waste management are not exempt from regulatory control because there are provisions that must be met appropriate legislation. In this case, the ICSD is equipment containing radionuclides have deliberately been incorporated and comply with the ISO standards.

In developed countries that have implemented standards and safety culture, in fact that disused ICSD can be managed by two options: hazardous or radioactive channel. For example in the European Union, most member states manage disused ICSD through hazardous, another country through the radioactive waste and in Australia, the disused ICSD must be managed if the their number more than 10 products. Of course, such a situation makes BAPETEN should conduct a study to consider the radiation safety aspects of major concern.

The scope of the focus of this paper is the regulatory control of waste that is part of the utilization ICSD, which according to the Act. No. 10 Year 1997 on Nuclear, is the utilization of such activities, including import, transfer, export, production, use, transport and waste management. While the method of the review is a review of literature study.

The understanding of radioactive waste (based on Act No. 10 Year 1997) is radioactive substances and materials and equipment that has been exposed to radioactive material or a radioactive substance due to the operation of nuclear installations that are not used anymore.While responsibility of hazardous is belongs to Ministry of Environment (KLH) andthe understanding of the hazardous waste (based on GR No. 18 Year 1999) is the residue of a business and/or activities that had the dangerous and/or poisonous due to the nature and/or concentration and/or amount, either directly or indirectly, can pollute and/or damage the environment, and/or can harm the environment, health, human survival as well as other living creatures.

If simplified, the definition of radioactive waste associated with radioactive ICSD is not used anymore. However, if it is described again, the radioactive substances are not used (disused source) is sealed sources that are not used (disused sealed radioactive source) or going to waste. Then going to waste (disused source) will turn into radioactive waste (spent source), which is sealed sources of radioactive waste.While understanding the related hazardousICSD is an activity that contain residual hazardous because of the nature of the concentration and the amount that can directly harm the environment, health, survival of humans and other living creatures. However, to ensure the waste management options ICSD track requires a scientific technical study on which the law set out in detail in the CR of BAPETEN.

Philosophy or principles of radiation protection in the use of the first ICSD is justified, the benefits must be more than the harm. Therefore, ICSDthat is used as a fire alarm equipment containing artificial radioactive substances, Am-241 sealed sources that are in accordance with ISO 9978 and tested according to ISO 2919 because the level of activity or activity concentration above the exemption level.

In all normal use, it will not cause radiation exposure exceeding the dose limit for members of the public values so that the use of ICSD does not harmful to health. Even in the event of fire, the ICSD really guaranteed safe or safe for the public. Safety provisions concerning the transport of radioactive materials require supervision but specifically regulated. Regarding waste manage-ment, are generally regarded as hazardous waste. Depending on the number and activity of radioactive substances. For example, in Australia, the number of ICSD more than 10 units must be collected and managed as radioactive waste.

The use of nuclear power as a consumer product in this industry should be controlled according to laws and regulations and harmonized internationally. The main concern of the radiation safety aspects of monitoring the ICSD as a product or goods circulating in the wider community is exposure to the public, which is regulated in the Government Regulation (GR) No.33 Year 2007 on Ionizing Radiation Safety and Security of Radioactive Sources. Similarly, the radioactive waste is also a regulated public exposure to the GR No 27 Year 2002 on Radioactive Waste Management. Recently, GR No. 27 is being amended and its current status in the process to be signed by the President of the Republic of Indonesia upon completion discussed in the Ministry of Justice and Human Rights.

Based on the data BAPETEN Licensing and Inspection System (B@lis), utilization license of the nuclear power as ICSD isnot exist yet in Indonesia. The problem is, there are only a few people who understand that there is a consumer productswhich containt radioactive and the consumerproducts is controlled by BAPETEN. For example, in two in recent years, there is a type of other consumerproducts, namely the lamps containing radioactive substances (high intensity discharge-HID lamps) have been get license from BAPETEN. Licensing process went after the employers' association lamps luminer reported this matter to BAPETEN to please policy aspects of radiationn safety. In fact, when it is reviewed from radiation safety aspects related to the potential exposure that was received by members of the public, so potential risks arise from ICSD is higher than HID lamps.

Information from the BATAN, there are several office buildings, industrial and mining are fitted with ICSD as a fire alarm. In fact, one of the largest mining companies in Indonesia, namely PT. Freeport has 1,800 units ICSDinstalled in the mine area Tembagapura, Papua. There is also information on running text on TV One, 22 to 23 February 2012 which proclaim that "as many as 3,033 fire alarms installed in Jakarta", however, does not mention the fire alarm containing radioactive.

Results and Discussions

Description and Working Principle of ICSD

The use of ICSD are installed in various buildings, such as offices, hotels and apartments has proven its superiority as a fire alarm that can save lives and property. This ICSD is very popular, because the goods are not expensive and sensitive to range wider fire conditions. In general, the amount of radioactive which used in ICSD is small, its range about 0.25 g of americium (Am-241) in the form of americium dioxide (AmO_2), which is equivalent to 30 kilobequerel (30 kBq). However, there are also ICSD containing radioactive substances whose activity is quite large. Therefore, this ICSD divided into two categories based on the value of activities, including:

a. ICSD containing Am-241, $\leq$ 40 kBq activity.
b. ICSD containing Am-241, the activity of > 40 kBq up to 2.2 MBq.

For certain types of fire alarm, ICSD containing radioactive substances, perform better than the conventional type of fire alarm.

Americium has the atomic number 95 and the average mass number 243. Americum are metals such as silver metallic, stained slowly in the air and can be dissolved in acid. There are no stable isotopes of americium, therefore it is very rare in nature. And there are 13 isotopes, isotope Am-243 is a half-life of more than 7,500 years and Am-241 a half-life 470

years. Alpha particles do not have enough energy to be able to cause damage to the skin, the penetration is very small with about 70 micrometers thick.

ICSD containing of radioactive substances and to be incorporated in ionization chamber called "ionization smoke alarm room". Radioactive substances in the form of a matrix amo2 packaged in gold and silver foil wrapped. The foil is thin enough so that allows alpha particles to penetrate and get into the ionisation chamber.

Alpha particles propagate only a few centimeters in air before alpha particle is absorbed so it will not be separated from the ICSD. Alpha particle is absorbed in the ICSD but most of the emitted gamma rays, but the percentage is small and its energy is also relatively small. Alpha particles interact with the room air ionization produces charged particles, called ions. Low-level electrical voltage is used to collect the ions inside the room, which causes electric current to flow. Smoke or hot air into the room which resulted in changes in the rate of ionization and the electrical current, which triggers the alarm. The physical appearance of ICSD if the lid is opened so that the visible signs of radiation on components containing radioactive Am-241, is given in Figure 1.

There are a number of old products from the ICSD using krypton (Kr-85), plutonium (Pu-238) or Pu-239 and Ra-226 which emits a very large external radiation compared to Am-241. ICSD the equipment is not used anymore (not justify) because it does not meet the radiation safety aspects, such as limiting the dose (dose constraint), namely: (1) an effective dose of the users and non-users in the use and disposal beyond the normal 10 mikrosievert (μSv) per year, and in the case of exposure to body parts, 1% of the dose limit for members of the public organs., and (2) doses of the users and non-users of

Figure 1. ICSD Containing Am-241.

ICSD arising from accidents are expected and reasonable and the error should not exceed the value of the use of members of the public dose limit 1 milisievert (mSv) per year. In fact, older products are not required to be labeled on the sign of radiation in the ICSD, as ICSD containing Ra-226, the arrow shows posis sources, is given in Figure 2. Label of radiation is one elements of the implementation of a safety culture.

Technical Requirements of ICSD

Am-241 activity values contained in the ICSD, that is ≤ 40 kBq sd 2.2 MBq much greater than the exemption level, ie 1 x 10^4 Bq or 10 kBq. Therefore, in order to ICSD can be marketed as a product or consumer goods, the safety requirements must be met, as follows:

- each sealed sources shall be in accordance with ISO 9978;
- activity or activity concentration must be below that achievable and reasonable satisfaction for the function;

Figure 2. ICSD Containing Ra-226.

- average dose rate covers an area of 10 cm^2 of each region can be achieved should not exceed 1 μSv/h from the surface of the ICSD;
- the product must be designed to prevent easy access to the radioactive source components, and
- the outer container must be designed to prevent the cracks as far as practicable.

Compliance with ISO 9978 for sealed sources should be followed up also adherence to ISO 2919 is applied during prototype testing ICSD. Tests carried out in order to meet the quality of products, include: (1) the test temperature (refrigerated at a temperature of-25^{0}C and heated at a temperature of 100^{0}C for one hour), (2) the impact test, (3) drop test, (4) the vibration test, and evaluation plus more specific test, namely: to test the effects of fire (ICSD must be heated from room temperature up to 600^{0}C) and incineration test (ICSD must be heated from room temperature up to 1,200^{0}C).

Labeling of ICSD

Any ICSD that will be marketed to the public must be equipped with a radioactive label, is given in Figure 3.

ICSD labeling provisions, as follows:

Sentence by writing the following: "WARNING-RADIOACTIVE MATERIAL";

- sign of radiation, and
- the identity and activity of radioactive substances contained.

Gamma dose rate emitted by ICSD at a distance of 1 m value is very small, which is a thousandth of the value of natural background radiation (BG). If the ICSD is held, the dose rate will be higher but still very small value, ie one tenth

Figure 3. Radioactive Label

of the value of BG. Because the hand is not overly sensitive to radiation when compared with other organs, such as the effects of gonadal organ radiation received by hand is not significant.

For officers who do not need to put ICSD there is concern about the excessive amount of radiation dose received by the installation of officers. Radioactive substances contained in the ICSD completely insoluble, if it is ingested radioactive substances radioactive substances are not absorbed in the body but are excreted in the digestive system.

In case of fire where fitted ICSD, the temperature may be more than 1,200 ^{0}C. When the temperature is so high, it may make radioactive sources radioactive source to melt but will not evaporate which resulted in internal radiation hazard by inhalation.

Waste Management Provisions of ICSD

In Indonesia, waste management approach to supervision of ICSD into a little dilemma, whether managed as radioactive waste or as hazardous waste. Moreover, if considered applicable regulations in developed countries, as follows:

Provisions of ICSD Waste Management in the European Union

ICSD waste management measures in the member states and candidate member of the EU can be made with a choice of 2 options, which is administered through the hazardous or radioactive waste, so depending on each country. Moreover, given the number by 15 European Union member states and candidate European Union member states also 15. Some countries require ICSD returned to the supplier or manufacturing after use, other countries consider that the waste management oversight ICSD out of control when it is sold, therefore ICSD waste can be managed with other domestic waste. The last European Commission Directive on waste electrical and electronic equipment (known as the WEEE direcrtive) requires that waste electrical products and electronic components containing radioactive collected separately and each radioactive component greater than the level of exclusion separated. Therefore, it is recommended that a mechanism for collecting, recycling and disposal is controlled according to type these wastes.

Waste Management Policy of ICSD in Australia

ICSD that is not used anymore will be a waste and ICSD is generally managed as hazardous

waste. In November 2001, the Australian Radiation Health Committee stated that the ICSD in small quantities managed as waste hazardous. However, if the amount of ICSD more than 10 (ten) tool, then ICSD must be gathered together for the bulk disposal, which means all the ICSD processed as radioactive waste in a place called the landfil. Provisions on the management of radioactive waste must meet the requirements of the National Health and Medical Research Council's Code of Practice for the Near-Surface Disposal of Radioactive Waste in Australia (1992) (RHS35).

IAEA Guide

Basic Principles of Radioactive Waste Management

By the IAEA in a number of publications, including BSS 115 Year 1996 and GSR Part 3 Year 2012 explained that radioactive waste management activities is exposure to members of the public (public exposure). To control radiological hazards and non-radiological contained in radioactive waste, there are nine basic principles of radioactive waste management should be considered in an integrated, include:

- **Principle 1: Protection of human health**
  Radioactive waste shall be managed in such a way as to secure an acceptable level ofprotection for human health.
- **Principle 2: Protection of the environment**
  Radioactive waste shall be managed in such a way as to provide an acceptable levelof protection of the environment.
- **Principle 3: Protection beyond national borders**
  Radioactive waste shall be managed in such a way as to assure that possible effectson human health and the environment beyond national borders will be taken intoaccount.
- **Principle 4: Protection of future generations**
  Radioactive waste shall be managed in such a way that predicted impacts on thehealth of future generations will not be greater than relevant levels of impact that areacceptable today.
- **Principle 5: Burdens on future generations**
  Radioactive waste shall be managed in such a way that will not impose undueburdens on future generations.
- **Principle 6: National legal framework**
  Radioactive waste shall be managed within an appropriate national legal frameworkincluding clear allocation of responsibilities and provision for independent regulatoryfunctions.
- **Principle 7: Control of radioactive waste generation**
  Generation of radioactive waste shall be kept to the minimum practicable.
- **Principle 8: Radioactive waste generation and management interdependencies**Interdependencies among all steps in radioactive waste generation and managementshall be appropriately taken into account.
- **Principle 9: Safety of facilities**
  The safety of facilities for radioactive waste management shall be appropriatelyassured during their lifetime.

Basic principles of radioactive waste management is also applies to SSRW, for example ICSD, are given in Table 1.

This Table is a modification of the table Classification of Radioactive Waste, IAEA Safety Series No. GSG-1 in 2009 so in accordance with the classification criteria outlined in the chapter on the amendment of Radioactive Waste Classification in GR No. 27 Year 2002. Especially for Cs-137, although half-life a little over 30 years, but is included in the criteria under 30 years.

The above table vii and viii numbers indicated that the Am-241 radioactive source is not ICSD but to use other types of consumer products, namely static eliminator and gauging. Radionuclides are used to ICSD is Am-241, a half-life 470 years and activity is $\leq$ 40 kBq. Thus, the waste included in the numbers viii ICD with a half-life criteria of > 30 years and activity is$<10^{10}$ Bq.

So that the overall basic principles of radioactive waste management can be applied in a manner consistent provision for all stages of activity, (starting from the collection, classification, processing, transport and/or disposal), the radioactive waste should be classified with a method.

Classification of Radioactive Waste.

The classification of radioactive waste can differ from one country to another as long as it is based on the basic principles of waste management. In Law. 10 Year 1997 on Nuclear stipulated that the classification of radioactive waste is divided into three types, include:

1. low-level waste (LLW);
2. medium-level waste (ILW), and
3. high-level waste (HLW)

The three types of this waste classification already includes the entire spectrum of general waste, including waste disposal ICSD. In order to be harmonic with IAEA recommendations and in line with the basic principles of radioactive waste management, then a change in classification only for LLW is divided into three levels, consisting of :

Table 1. Example of Classification of SSRW.

No.	Half-Life	Activity (Bq)	Volume	Examples
i.	< 150 days	10^8	Small	Y-90, Au-198, (brachytherapy)
ii.		5.10^{12}	Small	Ir-192 (brachytherapy) Ir-192, Se-79 (industrial radiography)
iii.	< 15 years	$< 10^7$	Small	Co-60, H-3 (tritium targets), Kr-85
iv.		$< 10^{14}$	Small	Co-60 (irradiator, radiotherapy)
v.	< 30 years	$< 10^6$	Small	Cs-137 (brachyterapy, moisture density detector)
vi.		$\approx 10^{15}$	Small	Cs-137 (irradiator) Sr-190 (thickness gauges, radioisope thermoelectric generators-RTGs)
vii.	> 30 years	$< 4.10^7$	Small	Pu, Am, Ra (static eliminator)
viii		$< 10^{10}$	Small, but may be large numbers of sources (up to tens of thousands)	Am-241, Ra-226 (gauges)

a. very short-lived waste (VSLW);
b. very low-level waste (VLLW), and
c. relatively low-level waste (RLLW).

ICSD waste management will depend on the determination of classification of radioactive waste for disposal later determined criteria.

Radioactive Waste Classification Methods

The IAEA provides several alternative parameters as criteria grouping waste into one of the five levels klassifikai SSRW. Based on a study by a team of amendments to CR No. 27 Year 2002 on Radioactive Waste Management of the literature IAEA Safety Standards, Classification of Radioactive Waste, No. GSG-1. There are two main parameters as quantitative criteria for the classification of radioactive waste, namely: a half-life ($T_{1/2}$) and the activity of radionuclide, which includes the activity concentration (A_c), total activity (A_t) and specific activity (A_{sp}).

The IAEA does not explicitly outline concerning the application of both the quantitative parameter to a method of classification. Quantitative approach of the IAEA only presented in graphical form "activity concentrations vs time half", is given in Figure 1. Both coordinate axes measured in a logarithmic scale and is only marked with a reference value to assist in interpreting the chart.
Note:

- TK: Tingkat Klierens (CL: Clearance Level)
- hari: day
- tahun: year
- Kandungan Aktivitas: Activity Content.

The Figure 4 is a modification of the FIG III-1 Classification of Radioactive Waste, IAEA Safety Series No. GSG-1 in 2009 so in accordance with the classification criteria outlined in the chapter on the amendment of Radioactive Waste Classification in GR No. 27 Year 2002.

From the description in the reference can be concluded that there is no quantitative limit values (activity and half-life) which can be generically applicable to all models of waste classification. Precise determination of the boundary values submitted to member states seem to be tailored to the capabilities and priorities of each negara.Telah determined that the values by interpreting graphs based on the origin and type of waste, radioactive waste classification method is obtained. In this case, LRST special classification (eg, Gauging equipment, Well Logging, ICSD), special low-level waste, is divided into three levels, as follows:

a. VSLW, restrictions apply to LRST with a half-life of more than or equal to 150 days;
b. VLLW, applies to SSRW with restrictions:

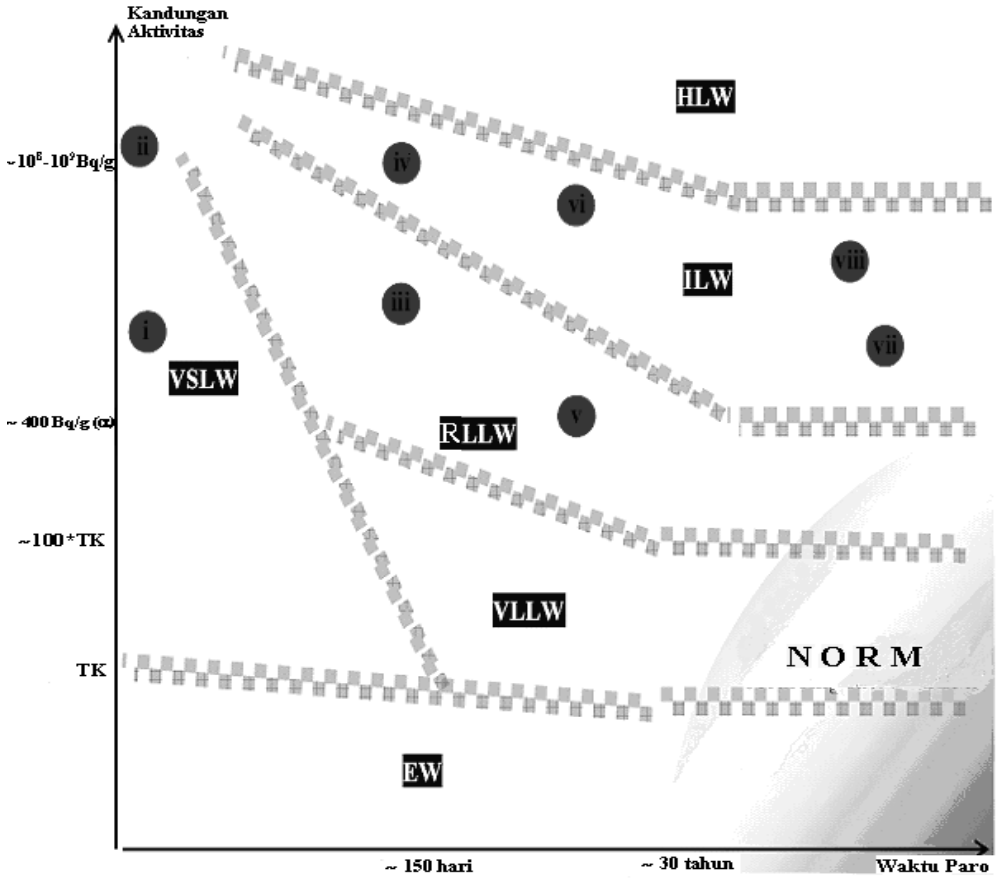

Figure 4. Graphic Activity Vs Half-Life.

i. if a half-life of more than 15 years, the activity limit of 10^5 Bq greater but less than 10^7 Bq, and
ii. if a half-life of more than 15 years but less than 30 years, the activity limit of 10^4 Bq greater but less than 10^5 Bq;

c. RLLW, applies to SSRW with the following restrictions:
i. if a half-life of more than 15 years, then the restriction activity of 10^7 Bq greater but less than 10^8 Bq; or
ii. if a half-life of more than 15 years but less than 30 years, then the restriction activity of 10^5 Bq greater but less than 10^6 Bq;

Disposal of Radioactive Waste

As set out in CR. No. 27 Year 2002 that, any person or entity who will perform the utilization of nuclear energy shall declare to BAPETEN that radioactive waste will be returned to the country of origin or submitted to BATAN to be managed. Thus, waste generators must refrain ICSD stages of waste management, especially with the intention of doing clearance carelessly discarded into the environment. although the activity of Am-241 is very-very small.

Given the user is not a licensee utilization of nuclear energy (use license), then the waste is the responsibility of the ICSD as a supplier of nuclear power license (import and transfer license) of BAPETEN. Therefore, the supplier is also waste generators must collect ICSD from the former part of the user to further comply with its obligations under the legislation. In this case, ICSD the former returned to the country of origin or stored in PTLR-BATAN, Serpong, Banten Province.

Although Australia set a limit to the number, the more dari10 (ten) ICSD the former to be managed through the radioactive waste, but that provision, perhaps less appropriate to be applied in Indonesia. Thus, the provision of waste management, the amount of ICSD the former need not be restricted. Moreover radiation sign attached to the ICSD will create a sense of fear that the psychological impact of public panic. In addition, the main consideration is the issue of safety culture and level of understanding of radiation safety is not the same between the Australia and Indonesia. In fact, if further review, the safety aspects of

radioactive sources, then the parties are likely to take advantage of bad faith for the purposes of terror conditions.

ICSD the left to PTLR-BATAN will be his responsibility to then do pre-disposal action finally disposal. After going through temporary storage, waste ICSD can be reclassified by the method of classification. Based on IAEA recommendations and assessment teams amendment GR No. 27 Year 2002, then PTLR-BATAN will keep ICSD in a landfill waste disposal depth is quite close to the soil surface (less than 5 meters).

Until now, the landfill facility for disposal of SSRW, including LLW does not exist. As one of the radioactive waste management facility is owned PTLR-BATAN interim storage of radioactive waste are also controlled by BAPETEN, is given in Figure 3.

Figure 5. Interim Storage of Radioactive Waste Facility as Downhole.

Location of interim storage facility located at the rear of Building 50 is the Head Office of Radioactive Waste Technology Centre and his staff, are given in Figure 6.

Figure 6. Building 50, Head Office of Radioactive Waste Technology Centre of BATAN.

Conclusions

1. In order CR of BAPETEN on consumer products more detailed set of technical requirements related to waste management can ICSD published in the near future so that the radiation safety oversight approach towards members of the public and the environment is more secured.
2. ICSD that can be imported and used in Indonesia should be limited to the type and content of radionuclide activity, namely Am-241 and a maximum value of 40 kBq activity and in accordance with ISO 9978 and ISO 2919 testing standards.
3. ICSD waste should not be disposed of carelessly let alone a radiation sign mounted on the tool can cause feelings of "unsafe/safe". Moreover, a large amount of waste in the form of ICSD must be managed through the bulk of radioactive waste and the amount does not need to be limited.
4. Based on IAEA guidelines, thusICSD waste is RLLW and stored in the landfill waste disposal depth is quite close to the surface (less than 5 meters).
5. BATAN already set on waste management rates of ICSD in CR. No. 29 Year 2011. Therefore, coordination among agencies authorized (BAPETEN and BATAN) regarding waste management continues to be improved so that the ICSD supervision of consumer products in general and especially to ICSD on appropriate national legislation and international harmony.

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Comparison of Patient Dosimetry Using TRS 457 Calculation and TLD-100 Measurement

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Abstract. **During recent years, patient dose has become a major issue and there is increasing awareness and greater realisation of the effects of ionizing radiation to human. Research on patient dosimetry was conducted to compare the entrance surface dose (ESD) measured using TLD-100 with the dose calculated using TRS 457 method. This research was done to know the difference of doses between direct measurement and by estimation, so finally it can be decided whether the latter can be used as an alternative method. The experiment initiated by ESD's measurement using acrylic phantom. And then, patient dose directly measurement performed by placing TLD-100 on the surface of patient's body whereas dose calculated from X-ray output using TRS 457. The result showed that the diffference of both dose determination techniques was less than 15%. Based on this result, TRS 457 dose calculation can be considered as an alternative to dose determination without performing measurements using TLD-100.**

Keywords: patient dosimetry, ESD, TRS 457, TLD 100

Introduction

The biggest contribution of radiation dose received by population of the world comes from the application of nuclear technique in the medical field especially from diagnostic X-rays which contribute more than 90% (UNSCEAR, 2008). It was due to the high number of examinations performed every year such as development of X-ray diagnostic equipment. Radiodiagnostic facility helps the doctor to diagnose the patient's illness. For conventional diagnostics, the radiation dose received by a patient is relatively low, however for the safety in its utilization it is important to pay attention in the protection of patients against radiation hazards, so the benefits of radiation is greater than its risk

Every source of radiation has the potential hazard for both radiation workers, patients and the surrounding community. Some adverse effects may occur in the human body when receiving excessive radiation. Adverse effects include hair loss, skin damage and disruption of the reproductive organs. The two basic principles of radiation protection of the patient as recommended by the ICRP are justification of practice and optimization of protection (ICRP, 1991; ICRP, 1996). Quality and safety have become hallmarks for efficient and successful medical intervention. Therefore, the measurement of patient dose need to be done. Currently, the patient dose measurement in diagnostic examinations is performed using TL LiF:Ti (TLD-100) dosimeter and read by Harshaw 2000A+B. However, due to the limited number of TLD-100 usage for any other research, it should finding an alternative dosimetry method. One is the dose measurement technique as described in TRS 457 which use of X-ray tube output to calculate the incidence air kerma. This research was conducted to compare the ESD of patient received using TLD-100 to the estimated dose using TRS 457 calculation method.

Materials and Methods

Before patient dose measurement using TLD-100 was carried out, the initial experiment was done using acrylic phantom (size 30x30 cm). Compliance test was applied to X-ray machine to know the equipment performance and to obtain the X-ray machine output data. TLD-100 was placed on the surface of phantom at it's central focus and then exposed with various condition according to the examinations that commonly used in daily practice. Three TLD's chip in a single package were used for each point of exposure. TLD-100 which has read using TLD Reader Harshaw Model 2000 AB evaluated using equation (1):

$$ESD = R_{average} * N_k * kQu \qquad (1)$$

where ESD is entrance surface dose (mGy), $R_{average}$ is average value of TL intensity from three TLD-100 chips after substracted by background value (nC), Nk is Calibration factor of TLD-100 on reference tube potential X-ray (mGy/nC) and kQu is correction factor of TLD energy dependence (IAEA - TECDOC, 2004 ; IAEA-

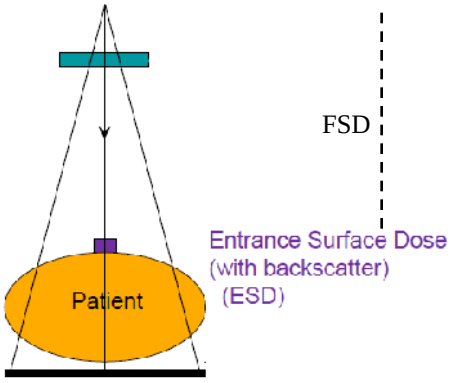

Figure 1. Entrance surface dose measurement

TECDOC, 2007). TL dose evaluation results were compared with K_i calculation refers to Code of Practice TRS 457.

Direct measurement of ESD of patient using TLD-100 was performed in several hospitals and adult patients were selected, which uses wide fields such as thorax, abdomen, vertebra and head. TLD used in this measurement had been grouped and calibrated such as TLD used in phantom experiment. Patient who enrolled in examination should be completed with recorded data included patient data (sex, age, weight and type of examination) and exposure parameters (kV, mA, s, mAs, FSD and field size).

Compliance test was also conducted for each X-ray tube which used in each Hospital in order to ensure the equipment compliance and to obtain X-ray tubes output data. Data from patient exposure condition and X-ray tubes output which used in measurement can be used to calculate incident air kerma K_i (mGy) according to TRS 457 as the following equation (2):

$$K_i = K_{a(V)100} * P_{it} * (100/FSD)^2 * BSF \qquad (2)$$

where K_i is incident air kerma or ESD, $K_{a(V)100}$ is X-ray tube output in air at 100 cm fokcus-surface distance and different voltage V (kV), P_{it} is load of tube which used in exposure (mA*s), FSD is Focus Source Distance, the distance of source to patient's skin/surface (cm) and BSF is Back Scatter Factor of exposed object (IAEA-TECDOC, 2004; IAEA-TECDOC, 2007).

Backscattering factor (BSF) value which used in this calculation refers to Appendix VIII of TRS 457 with assumed that field size was 25x25 cm and the patient's body equal with acrylic phantom.

Results and Discussion

From the experiment using phantom which had been conducted, the X-ray machine performance result was still complied. This was indicated by parameters which consist of kV and timer accuracy, reproducibility (kV, timer and dose), beam quality and linearity of output that was still within the tolerance limit. Therefore, the X-ray machine can provide the reliable measurement result. From compliance test, it can be determined output curve as shown figure 2.

ESDs on the phantom with TLD-100 were calculated using equation (1) whereas the incident air kerma dose that refers to E_{SAK} of TRS 457 obtained from X-ray machine output data was calculated using equation (2). The difference of those both results was ±6%, as shown in Table 1. In general, the incident air kerma dose that calculated was greater than that ESD from TLD-100, but both were relatively similar.

The difference of ESD above was caused by X-ray beam quality that used in this experiment which was less than that described in BSF table on TRS 457.

Refer to the results in Table 1, the experiment was continued with direct measurement in patients according to the planning. The results of patient dose measurement using TLD-100 and incident air kerma dose calculation using TRS 457 are shown in Table 2. The data shown in Table 2 obtained from patients dose measurement in four hospitals using four different X-ray machines which passed the compliance test. There are four kinds of patient

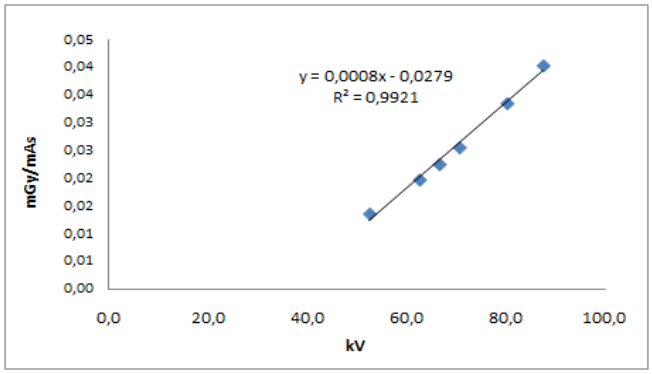

Figure 2. X-ray tube output curve

Table 1. Comparison of ESD measurement from TLD-100 on phantom with calculation using TRS 457.

kV	mA*s	*)ESD_{TLD} phantom	*)$ESAK_{TRS\ 457}$	$ESD_{TLD}/ESAK_{TRS\ 457}$
60	20	0.7007	0.7100	0.9869
60	20	0.6817	0.7100	0.9601
50	16	0.3951	0.4097	0.9643
71	15	0.7555	0.7303	1.0346
55	12	0.3408	0.3637	0.9372
*) in mGy				

Table 2. Comparison of ESD patient measurement using TLD-100 with calculation using TRS 457

Examination	$ESD_{Patient}$ *)			$ESAK_{TRS\ 457}$ *)			$ESD_{patient}/ESAK_{TRS}$ (Mean)
	Mean	Min	Max	Mean	Min	Max	
Abdomen	1.3127	0.5568	1.9470	1.1650	0.5901	1.6164	1.1268
Head	0.8548	0.5003	1.0869	0.8264	0.5900	1.3499	1.0344
Thorax	0.3777	0.0168	0.9969	0.3505	0.0216	0.8336	1.0776
Vertebra	1.9849	0.1092	6.1458	1.9573	0.1111	6.4527	1.0141

*) in mGy

examination that was observed. The mean ESD obtained from this experiment was still within the limit of Dose Guidance prevails in Indonesia (IAEA-TECDOC, 2004; Perka BAPETEN, 2011). However, based on the result, it was only the thorax examination which showed fairly wide dose range (Table 3). Different from the test conducted on acrylic phantom, the patients dose measured by TLD-100 showed a greater trend than the dose calculated using equation (2). The maximum difference of the mean value of the second dose was less than 15%. Illustration of the difference in these value's is presented in Figure 3.

Table 3. The level of radiology diagnostic dose guidance for adult patient compare with patient dose measurement using TLD-100.

No.	Examination	Projection	$ESD_{Guidance}$ mGy	$ESD_{Patient}$ mGy
1	Thorax	AP / PA	0.4	0.3777 ± 0.3191
2	Abdomen	AP	10	1.3127 ± 0.6126
3	Head	PA	5	0.8548 ± 0.2302
4	Cervical/ thoracal	AP	7	1.9849 ± 1.9303

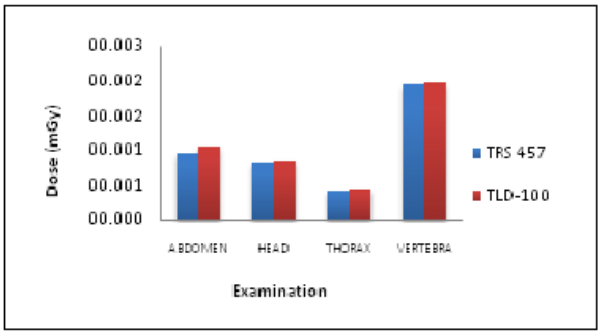

Figure 2. Dose Comparison of TRS 457 with TLD-100

The calculation of patient dose is influenced by setting conditions, such as kV, mAs and FSD. X-ray beam quality used in this experiment also significantly influence the dose because it associated with the backscattering factor (BSF) from exposure object.

Good radiological techniques as the application of high kV with other appropriate parameter settings will give optimum results in both imaging and low patient dose (Vassileva, 2004; Craig, 2000). Ideally, the technique used in the calculation of dose is using pda Code of Practice TRS 457, but the fact showed that all hospitals examined in this experiment do not match these conditions. For example, some hospitals are using exposure conditions on chest examination with voltage below 70 kV and less than 150 cm FSD.

In addition, the different of field size between patient dose measurements using TLD-100 to which was used in BSF determination refers to TRS 457 also contributed in the different results. BSF table refers to TRS 457 using field of 25x25 cm size, not as same as the real exposure field size that depend on the size of tape/film used in the hospital, which were 24x30 cm, 35x35 cm and 35x43 cm. As information, the dose measurements using phantom which conducted also using field of 25x25 cm size and the distance between focus to phantom surface is 100 cm.

Another factor which influenced the difference between dose measurement using TLD-100 and calculation using TRS 457 is the shape and size of the experimental room. Therefore, the backscatter value increased and it caused the greater dose received by TLD-100 whereas the shape and size of the room for TRS 457 reference is the ideal room.

According to the describe above it can be understood if the dose measurements in phantom using TLD-100 provide similar results to the calculation of X-ray output data refers to TRS 457 compared to the direct dose measurement using TLD-100 in patients.

Conclusion

Comparison of results of dose measurements using TLD-100 to phantom dose calculations using the TRS 457 showed almost similar results. Dose difference between the two methods was ± 6%. Comparison of dose measurements using TLD-100 in patients with dose calculations using the TRS 457 showed the differences of <15%. Based on these experimental results, the TRS 457 dose calculation can be considered as an alternative dose determination of patients without performing measurements using TLD-100.

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Influence of Safety Culture Model Towards OHS Behavior of Radiology Services of X Hospital

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Abstract. **Safety culture plays an important role in shaping occupational health and safety (OHS) behavior on radiology services in hospital radiology installations. Thus efforts to reduce the negative effect on radiological installations should begin by forming a good safety culture. However, studies on safety culture and its influence on OHS behavior in a hospital radiology service had never been there. This study aimed to determine empirically the effect of safety culture on OHS behavior. To achieve this goal, first the model is hypothesized consists of six safety culture factors and one OHS behavior. Then the model is evaluated using partial least square- structural equation modeling (PLS SEM). Data obtained by spreading 60 set of questionnaire to 73 workers of Radiology Installation of X Hospital whose have a record of radiation doses. Thirty eight sets of questionnaire gathered and used to evaluate the hypothesized model. The evaluation result shows that only four-factor model of safety culture that provides significant influence on OHS behavior. This paper concludes that: (1) commitment of top management are the key drivers of safety culture, and (2) to control OHS behavior through discretionary approach is the most important.**

Keywords: radiology, safety culture, OHS behavior, structural equation model, partial least square.

Introduction

At this moment, using of ionizing radiation has been growing rapidly in many fields of life, even for the field of health/medicine. Ionizing radiation used in the medical field can be either X-rays, γ-rays or other ionizing radiation.

Although it is benefits for humans, the potential dangers of ionizing radiation can't be ignored. This radiation hazards due to ionizing radiation through the material when the collision occurred photons with atoms of the material that will cause ionization. These incidents possible harm to the body, whether they are deterministic, or stochastic. This negative effect can be either acute somatic (burns, infertility, cataracts, etc.), somatic effects over time (late somatic effect) such as cancer and genetic effects (Wiharto and Budiantari, 1997).

To prevent the negative effects of radiation so made dose limit value (DLV), which is the largest value permitted radiation dose can be received by the employees and community members, particularly for patients to be applied optimization implemented by limiting the dose and the level of guidance for medical exposure (PP No. 33 Tahun 2007, Keputusan No. 01/Ka-BAPETEN/V-99, and Keputusan No. 01-P/Ka-BAPETEN/I-03), this philosophy is known as the concept of ALARA (As Low As Reasonably Achievable) (Cember, 1992). To effectively implement this philosophy needs to evaluate its implementation and identify the factors that influence it. For that we need continuous efforts to improve occupational health and safety (OHS) of this ionizing radiation, such as measuring the level of safety.

Recently, there has been a change or a trend in the measurement of occupational safety, from measurements the amount or rate of accidents become to measures that focus on culture (climate) safety (Cooper, 2000). The term safety culture first emerged in the OECD Nuclear Agency Report 1987, published in INSAG Year 1988 which was motivated by the nuclear reactor accident at Chernobyl in 1986 (Cooper, 2000). The term safety culture was then internationally understood as corporate culture or atmosphere in which safety issues are understood and accepted to be the main priority in the company.

Some definitions of safety culture of According to Utal (1983) and Turner (1992) in Andi et al. (2005), namely: first by Utal (1983) "Safety culture is the beliefs and values associated with control systems and organizational structures that form the norms of behavior". Second by Turner (1992) "safety culture is a series of beliefs, norms, behaviors, rules, and technical and social practices are closely related to efforts to minimize the dangers and accidents that will happen to workers, managers, clients, and society." Third, according to INSAG-4 (IAEA, 1991) "safety culture is a combination of characteristics and attitudes in organizations and individuals which establishes, as a priority, nuclear installation safety issues get attention in accordance with its interests. Therefore, safety culture has two main components. The first is the necessary framework within an organization, it is the responsibility of management. Second is the attitude/behavior of staff at all levels to respond to and utilize the framework.

Although safety culture is an abstract concept but plays an important role in determining the performance and environmental safety of workers in industries that use radiographic techniques with high radiation doses. This was identified in the RTD Netherlands. Therefore, safety culture should be developed and maintained by the company in promoting a positive work attitude to the safety of radiation (Van Sonsbeek, 2006).

According to Reason (1997) in Andi et al. (2005) two main causes of the failure of the safety system is not secure workers' behavior and latent conditions from organizational factors and work environment. Therefore, efforts to prevent accidents will be more successful if the management get rid of the problems that existed at the company as early as possible, that is organizational factors. In addition to removing the problem, this effort will form a good safety culture and to encourage workers to behave safely. Safe behavior is an important note in radiation facilities, because 68% of the causes of radiation accidents in the year 1960-1968 was operator error, as reported USEAC (Batan, 2006).

Radiation hazards are usually latent and long-term impact, so the effort to establish safe behavior, namely OHS behavior worth noting. as according to Reason, at each facility users of ionizing radiation or nuclear power required to achieve safety culture (PP No. 33 Tahun 2007).

Therefore, the model of safety culture influence on the OHS behavior is needed to understand the interaction between the factors forming the safety culture and its relationship with the OHS behavior. So the company can develop strategies to improve the OHS behavior for workers on an ongoing basis.

Base these problems, this study aims first to develop model of the relationship between safety culture and its factors with OHS behavioral of radiology workers who have a record of personal radiation doses. Second, to test the relationship between the culture of safety to OHS behavior, both direct and indirect relationships.

Methodology

The study begins by defining the safety culture factors and indicators, also OHS behavioral indicators. Determining factors and indicators is done by literature studies, field observation, and opinion of the competent parties.

In determining the safety culture factors, researchers refer to the six main factors forming the safety culture that is often used in industry (Cheyne et al., 1998, Oliver et al., 2002, and Mohammed, 2002 in Andi, et al., 2005), namely: (a) commitment of top management (2) rules and safety procedures (3) communication (4) the competence of workers (5) the involvement of workers and (6) work environment, each of which will be explained in the following paragraphs.

According INSAG-4 (IAEA, 1991), to establish a safety culture must start from the beginning, namely top management, for example, top management makes policies that demonstrate its commitment to safety. Research conducted Cheyne et al. (1998), Mohamed (2002) in Andi et al., 2005 and Mars (1998) in Andi (2008) showed that the factor of commitment of top management are key success factors form the safety culture. Top management commitment can be a concern for safety, proactive to anticipate dangers that may arise, and reactive when the accident occurred. This proactive action can be a complement of workers with personal protective equipment, being reactive actions can be brought to the hospital when the accident occurred.

Regulations and safety procedures is one factor to minimize accidents, because it can give an illustration/restrictions clearly in the implementation of safety programs (Pipitsupaphol, 2003 in Brenda, 2008). In INSAG-4 (IAEA, 1991) stated that the regulations and safety procedures that should contain the responsibility, through formal assignment, job descriptions and understanding by the individual. But the rules and procedures were often not in accordance with the actual working conditions, difficult to be conducted, or lack of benefits, so that it encourages employees violate safety rules and procedures that exist. For example, based on the

writer's observation in the Radiology Department, its workers often do not use the recorder of radiation dose when working, because it is not an effective tool for the existing conditions.

Safety programs should be supported by good information management system, namely the path of good information from management to workers or the reverse of the workers about unsafe conditions to management, as stated by Davies et al. (2001), Tony (2004) in Andi et al. (2005). The latest information is very important, especially regarding the latest safety regulations, and subject to hazards in the workplace. INSAG-4 (IAEA, 1991) states that communication also needs to be built through supervision, including audit and review activities, with a readiness to respond to questioning the attitude of the individual and the curiosity of every individual. This attitude raises awareness of the dangers that may arise.

Mohamed (2002) in Andi (2008) describes competence as the knowledge, understanding, and responsibility of workers to jobs, as well as knowledge of the risks and dangers that threaten the workers on the job. This competency is primarily acquired through the training and instruction of personnel and taught myself. Therefore, in the form of competence, the management is very influential.

A good working environment makes workers feel secure and not awkward in doing his job. INSAG-4 (IAEA, 1991) states that in order to create a work environment that supports the culture of safety needs to be invested awareness of the importance of individual salvation, and the commitment, the demanding role models at senior management level in prioritizing safety, and adoption by individuals about the purpose of public safety .

Cheyne et al. (1998) in Andi et al. (2005) and INSAG-4 (IAEA, 1991) found that worker participation in safety programs is very important as an awareness of safety programs Involving workers to grow the safety culture need motivation through leadership, goal setting, reward systems, and sanctions, and through attitudes individual consciousness. Indicators of the six safety culture factors that have been described before are presented in Table 1.

Table 1. Safety culture and OHS behavioral factors with their indicators.

Code	Factors & Indicators
A	**Commitment Top Management**
A.1	The hospital is concerned about the safety
A.2	The hospital will stop the activities/jobs that endanger safety
A.3	There are efforts to improve safety performance at a certain period
A.4	There is supervision on safety
A.5	Hospitals provide safety equipment
A.6	The hospital provides comprehensive training to workers about the OHS
A.7	Management notices workers opinion before making decisions relating to safety
A.8	Management maintains an open policy regarding safety issues
A.9	Corrective action is always carried out when the management informed of the practice/situation unsafe
A.10	Management provides a good safety equipment for workers, patients, and the environment
A.11	Safety becomes a primary consideration in allocation of resources (human resources, budget, etc.)
A.12	Management is always present in meetings to discuss safety
A.13	Safety performance indicators are always observed, evaluated, and acted upon by management
A.14	Management noticed quality assurance of the radiology services
B	**Regulation and Safety Procedures**
B.1	Rules and safety procedures is required
B.2	Easily implemented safety procedures on the job
B.3	There are penalties for violation of rules and safety procedures
B.4	Regulations and safety procedures are enforced hard by the management
B.5	Regulations and safety procedures are improved on a regular basis
B.6	Safety regulations and procedures are easy to understand
B.7	Rules and procedures are applied fairly to maintain the safety
B.8	Safety inspection (workers, patients, and the environment) run on a regular basis
B.9	Preventive maintenance carried out regularly in the workplace
B.10	No deviation from a safe working procedures and correctly even if pressed by time or felt it was unusual and could be.
B.11	Feeling easy to follow all the rules and safety procedures when working

Table 1. Continued.

B.12	Quality of rules, procedures and documentation are good
C	**Communication**
C.1	Satisfied with the submission of information about job
C.2	Feeling always get the latest information about safety
C.3	Established good communication between the workers and the managerial
C.4	Established good communication when replacement workers
C.5	Feeling informed of accidents or hazards that occur
C.6	There are enough opportunities to discuss and agree on safety issues at a meeting
C.7	The purpose and the target safety programs in the workplace was clearly
C.8	Feeling not understand what should be done and who should be known if any potential hazards in the workplace
C.9	The relationship between management and workers are built on the basis of mutual trust
C.10	Responsible for safety is very clear at all levels of management and each worker
C.11	Each X-ray machine or radiation source equipped accompanying document that provides a detailed explanation of the available protective equipment, so we can have safe conditions
D	**Competency**
D.1	Feeling understand that the responsibility for safety
D. 2	Feeling understand how to work safe on their job
D.3	Feeling fully understand the risk jobs
D.4	Education and training provides a clear understanding of the safety
D.5	Feeling never do work outside the responsibility
D.6	To refuse to perform dangerous jobs
D. 7	Feeling understand how to use safety equipment and understand standard operating procedures
D.8	Feeling understand how to maintain and improve safety and health of workplace
D.9	Feeling understand how to reduce the risk of work accidents
D.10	Feeling trained enough to respond to emergencies in the workplace
D.11	There are systematic development program to improve individual competence
D.12	Ability of leadership is well developed systematically
E.	**Work Environment**
E.1	The workers care about safety
E.2	There is no culture of blame when an accident
E.3	Job is not boring
E.4	Workers motivation increase by safety programs
E.5	Feeling satisfied with the work environment safety (safety equipment, cleanliness, bright atmosphere, etc.)
E.6	Feeling never get pressure on the job
E.7	There are significant dangers inherent in the workplace
E.8	In the workplace, feel the possibility of complexity arises extremely high when there is danger
E.9	Safety conscious behavior is supported by management and workers both in formal or non formal
E.10	The factors that support motivation and comfort of work are considered by management
E.11	Cooperation interdisciplinary, cross-functional, and team work can be done well
F	**Worker Involvement**
F.1	The management of workers involved in delivering information
F.2	Workers engaged in the development of safety procedures
F.3	Management appreciate workers who report any accidents / hazards to safety
F.4	Workers were asked to remind workers about the dangers to safety
F.5	Using all the necessary safety equipment to do the job
F.6	Following the rules and safety procedures correctly when running a job
F.7	The volunteer tried to raise the level of safety
F.8	Encourage coworkers to work safely

In setting the indicators of OHS behavior, researchers refer to two things, namely: (1) three-forming aspects of the behavior of OHS, ie aspects of discipline, caution, and hygienic (Abidin, 2007), (2) three principles of radiation protection and safety, namely justification, limitation, and optimization (Cember, 1992). OHS behavioral indicators are presented in Table 2.

Hypothesis of the influence model of safety culture on the OHS behavior in this study with previously established a relationship linking to illustrate the path of influence between the factors that have been described previously (Supranto, 2004). Determination of influence lines on the model hypothesis is based on a literature study has been done in previous research by Cheyne et al. (1998), Oliver et al. (2002) and Mohammed (2002) in Andi at al. (2005), also safety culture described in INSAG-4 (IAEA, 1991). Hypothesis of the model can be seen in figure 1.

Subjects of research were all employees of Radiology Department of X Hospital which has a record of radiation doses totaling 73 people. Data to test the hypothesis model is obtained by propagating 60 sets questionnaires to the employees.

Questionnaire consisted of three parts: respondent profiles, safety culture, and the OHS behavior. Respondents were asked to provide the level of agreement each statement/question using 5 Likert scale, namely 1 (extremely inappropriate or never) to 5 (very appropriate or always),.

The size of the sample as a requirement for testing by PLS is 30-100 respondents (Ghozali, 2008). Number of questionnaires collected was 38. Dissemination of questionnaires was carried out for approximately two months. Adequacy of sample size was calculated by formula $n = \left(\frac{\tau_{\alpha/2} s}{e} \right)$ (Walpole et al., 1998) with a significance level (α) 5%, percent of clearances (e) 3%, and standard deviation (s) 0.082, which is 29 respondents, the respondent of 38 was enough.

To evaluate the model in this study used SEM PLS, which consists of two evaluation stages, namely the outer evaluation model (reliability and validity test of data) and evaluation of the inner model (R^2 test, test of significance of path coefficients, and Q^2test). Evaluation conducted by SmartPLS software.

Table 2. Indicators of OHS behavioral.

Code	Factors & Indicators
G.	**OHS Behavior**
G.1	Using a film badge when working in radiated workplace
G.2	Wearing PPE according to the rules when doing radiology jobs
G.3	Preparing notes and fill in the log book when it will do the job radiology
G.4	Recording or reporting to the authorities when there are accidents radiology
G.5	Saving back into the place after wearing film badge
G.6	The radiologist doing the work necessary radiation protection officer
G.7	Environmental around the radiology workplace be done radiation monitoring
G.8	Closing the door when doing radiology jobs
G.9	Giving appropriate hazard warning sign when doing the work of radiology procedures
G.10	Setting up monitoring of doses that can give a warning with the sound of a combination of time, file size, and output radiation when doing the work of radiology/radiodiagnostic
G.11	Creating radiation beam size as small as possible as required examination before doing the work of radiology/radiodiagnostic
G.12	Trying to ask a competent person when in doubt in the work related to radiation safety
G.13	In using the measuring tool, try using a calibrated tool.
G.14	Often joking while doing in radiology jobs
G.15	Calculating the required size of dosage before doing radiology jobs
G.16	Regularly check your radiology equipment is still working normally
G.17	Managing patients with patience and careful so that the imagery well before doing radiation to patients
G.18	Answering patiently and clearly when the patient / family to ask something related to radiology (eg, asking about the doses used)
G.19	Although it is not radiological dirty after doing the job, keep washing your hands
G.20	Before eating / drinking, washing hands first
G.21	Maintain cleanliness of the body with a bath after work

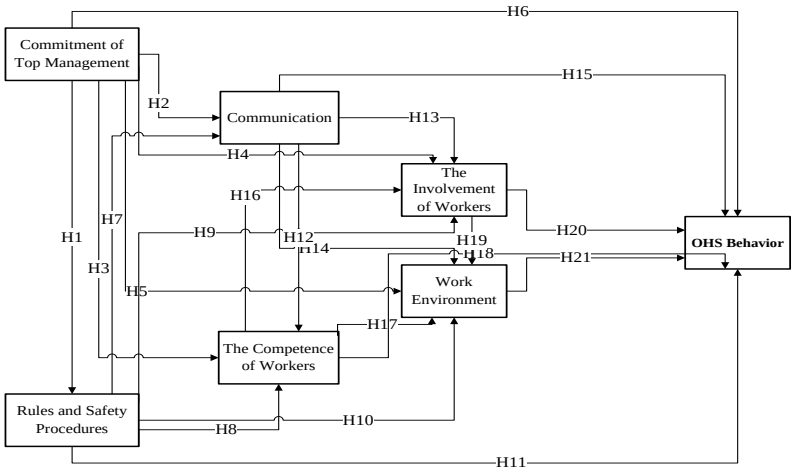

Figure 1. Hypothesis of relationship model of safety culture toward OHS behavior.

Results of evaluation

Validity testing consists of convergent validity and discriminate validity. Convergent validity with reflexive indicators assessed its loading factor, its value is recommended 50-60 (Chin, 1998 in Ghozali, 2008). Discriminate validity with reflexive indicators assessed by cross loading. Reliability testing using composite reliability, value 0.60 recommended. After four times the estimate derived factor/construct (and indicators) are reliable and valid the significance level (α) 5%, ie: A (the A.1, A.2, A.3, A.4, A.5, A .7, A.8, A.9, A.10, A.11, A.13, A.14), B (the B.3, B.4, B.5, B.7, B.8, B.9), C (C.3, C.5, C.6, C.7, C.8, C.10, C.11), D (D.1, D.2, D.3, D.4, D.7, D.8, D.9, D.11), E (E.1, E.2, E.4, E.5, E.9, E.11), F (F.1, F.2, F.3, F.4, F.5), G (G.5, G.11, G.12, G.13.G.15, G.16, G.17 , G.18, G.19)

Significance test for path coefficients do to evaluate the effect of 21 paths in the figure 1 with the significance level (α) 5%. There are 10 paths that are not significantly influenced, then the line is eliminated to form a new model (Figure 2).

Values of R^2 are 0.67, 0.33, and 0.19 for the endogenous variables indicate that the model of "good", "moderate", and "weak"(Ghozali, 2008). From R^2 test can be seen that models created in this study include the moderate model because the value of R^2 for the OHS behavior is 0.52. From Figure 2 enable to know the direct effect, indirect effect, and the overall effect as seen in Table 3.

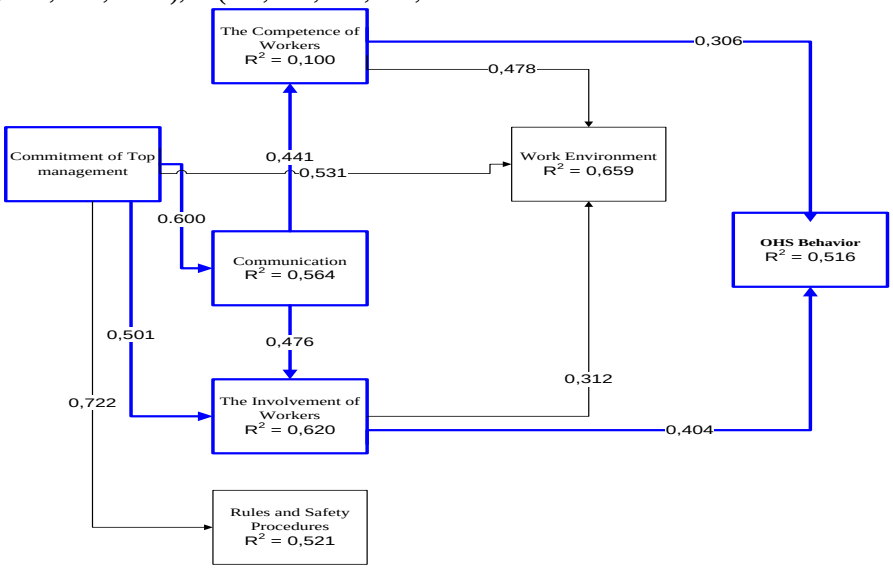

Figure 2. Improved model, the value of direct effect, and R^2.

Table 3. Influence of safety culture factors on the OHS behavior.

No.	Safety culture factors	Direct effect	Indirect effect	Comprehensive effect
1.	Commitments of top management	-	0,399	0,399
2.	Safety rules and procedures	-	-	-
3.	Communication	-	0,327	0,327
4.	Competency	0,306	-	0,306
5.	Work Environment	-	-	-
6.	Worker Involvement	0,404	-	0,404

The third test calculates Q^2 with R^2 of the endogenous variable (figure 2), therefore:

$$Q^2 = 1-(1-R_B^2)(1-R_C^2)(1-R_D^2)(1-R_E^2)(1-R_F^2)(1-R_G^2) = 0{,}988$$

Since Q^2>> 0, close to 1, then very good model based predictive relevance value.

Discussion

From Table 3 there are four factors of the safety culture that influence the OHS behavior , namely: top management commitment, communication, competence of workers, and worker involvement. The research shows that although top management commitment factor does not directly influence the OHS behavior, but the main factor in influencing culture safety. Regulations and safety procedures, and work environment had no effect.

Top management commitment are the main factors that in accordance with those of Turner (1992) and Pigeon (1998) in Andi et al. (2005) that the commitment of the management of both the form of actions, writings, and the words, become the most important factor for the creation of safety culture. Likewise INSAG-4 (IAEA, 1991) states that in order to establish safety culture must start from scratch, from top management and the success of their applications is reflected by the commitment of management and the competence of workers. Thus, top management should look at safety as an integral part of strategies to control the risk of radiation. Moreover, long-term radiation effects, easily overlooked.

According to the Reason (1997) in Andi et al. (2008) worker behavior can be controlled through a strict approach (prescriptive) based on rules, flexible (discretionary) based on experience/training, or a combination of these two approaches.

The results of this study indicate that the rules and work procedures have no effect on the OHS behavior. This is in accordance with the results of research by Yusri and Situmorang (2000) stating that the hospital radiation workers give bad feedback on the effectiveness of safety procedures and safety systems.

The model shows that OHS behavior more accurately controlled discretionary. Its reason is although nothing the direct influence of communication factors on the OHS behavior but the total effect is relatively large, also communication between staff and management are important factors to improve the competence and involvement of workers, where the two factors are very influential on the OHS behavior.

Discretionary control can be said to control internally, which in this study was to raise awareness of the importance of working within safety culture, which can be done by involving workers optimally, two-way communication with workers, providing education/training about OHS, protection and radiation safety, and other fields to improve the competence of workers. This is according to Keputusan No. 01-P/Ka-BAPETEN/I-03 BAPETEN Head, namely that associated with radiation workers must receive training or job-related effects or risks associated with the work.

Conclusions

This research shows that the safety culture influence the OHS behavior primarily through four factors, ie: commitment of top management, communication, competence, and involvement of workers.

To control the OHS behavior, especially those related to the safety of radiation more accurately through a discretionary approach.

Worker competence of radiological installation determine for radiation doses received by patients, the environment, or workers themselves. This

competence include technical competence, OHS, also radiation protection and safety.

Finally, researchers suggest the need to improve competence through education and training for workers, particularly problems of radiation protection and safety. Also more attention should be directed to the safety culture factors in shaping the OHS behavior. However, efforts to measure or evaluate these factors will not be easy. It is expected that future research provides input to evaluate the safety culture in hospital radiology installations.

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Assessment of Internal Dosimetry Due to the Administration of 99m-Tc Sestamibi in Nuclear Medicine

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Abstract. **Nuclear medicine in Indonesia has become one option in diagnostic procedures for detecting abnormalities of physiology and or pathophysiology in human body. The procedures have been done after the administration of a small amount of radiopharmaceutical into human body in order to gain the imaging features of organ targets. In this procedure, the administration of radiopharmaceutical has induced some organs receive radiation dose from target organs, and the body will get an internal radiation dose. The internal dose should be assessed regarding to radiation protection purposes. The aim of this study was to establish an image quantification method to asses internal dosimetry in each patient, due to the administration of 99m-Tc Sestamibi and to evaluate the distribution of radiation dose in the patients using MIRDOSE3, software from the United States Society of Nuclear Medicine. Approximately 370 MBq of 99m-Tc Sestamibi was administered into the patients and the patients were then scanned by using the dual-head gamma camera AnyScan MEDISO. The image quantification of organ was introduced by selecting the region of interest from each organ targets and continued by the quantification using the conjugate view method to get the activity and absorbed dose. The results were compared to the internal dose studies using MIRDOSE3. As a result, the image quantification of 20 patients were mostly inconsistent with the result from MIRDOSE3 calculation. It showed that every patient has specific dose distribution, which might reflect the variation of anatomical and physiological condition of each patient. It also showed that in further investigation, another correction method should be applied, and a time based investigation should be introduced. Hence, the quantification method in this study will be of benefit for evaluation of the procedures using 99m-Tc Sestamibi and other radiopharmaceuticals for other diagnostic procedures.**

Keywords**:** ***diagnostic procedures*****,** ***Internal Dose, 99m-Tc Sestamibi***

Introduction

Nuclear medicine in Indonesia has become one option in diagnostic procedures for detecting abnormalities of physiology and or patophysiology in human body. The procedures have been done by the administration of a small amount of radiopharmaceutical into human body in order to gain imaging features of organ targets. When the radiopharmaceutical reaches the organ targets, the radiopharmaceutical will be accumulated and released in particular ways. The accumulation of radiopharmaceutical will lead the organs become radioactive sources inside the body. Hence, the human body will get internal dose from the radioactive organs.

Regarding to radiation protection purposes, the administration of radiopharmaceutical into human body need to be assessed in order to consider the risk to the patients and the critical groups such as the family of patients, nurse and people who have contact with them (Helal, 2012). Moreover, the assessment might be useful for other purposes such as evaluation of clinical trials or internal dose assessment for new radiopharmaceuticals (Stabin et al., 1999).

One of well-known radiopharmaceutical for nuclear medicine procedure is technetium-99m 2-methoxyisobutyl isonitrile (99m-Tc-Sestamibi). 99m-Tc Sestamibi previously has been used for myocardial perfusion studies (Joseph et al., 2003), but the application has been extended in scintimammography in breast cancer detection (Koga O, 2010) and other sites such as thyroid (Perez-Monte, et al., 1996), brain tumors (Yokogami et al., 1998), and multiple myeloma (Pace, 2005). In this study, 99m-Tc Sestamibi was used for cardiac studies, whole body imaging, and breast scintimammography.

The main objective of this study is to establish an image quantification method to asses internal dosimetry in each patient, due to the administration of 99m-Tc Sestamibi, as a result of image acquisition using a dual-head gamma camera AnyScan MEDISO. The result of quantification will be compared to the dose calculation using MIRDOSE3, a software from the United States Society of Nuclear Medicine. The software needs some data input such as the biokinetic data of 99m-Tc Sestamibi, the phantom was used in the study, and the number of administered radiopharmaceutical. The biokinetic data of Sestamibi used as the residence time in the organs. However, only few organs has been chosen as the organ targets, namely liver, kidneys, heart, thyroid and bladder.

Internal radiation dose assessment

A generic equation in internal radiation dose studies has been acknowledged in few references

(Helal N, 2012; Stabin M. and Flux G., 2007; Sgouros G, 2005). The equation describes the absorbed dose rate in the organ as follow :

$$\dot{D} = \frac{k\,\tilde{A}\,\Sigma_i\; n_i E_i \phi_i}{m} \qquad (1)$$

where,
D = absorbed dose rate to a target region of interest (Gy/sec or rad/hr).
$\tilde{A}$ = activity (MBq or µCi) in source region.
n_i = number of radiations with energy Ei emitted per nuclear transition.
E_i = energy per radiation for the ith radiation (MeV).
ϕ_i = fraction of energy emitted in a source region that is absorbed in a target region.
m = mass of the target region (kg or g).
k = proportionality constant (Gy-kg/MBq-sec- MeV or rad-g/µCi-hr-MeV).

However, the United Stated Society of Nuclear Medicine has established the Medical Internal Radiation Dose (MIRD) method which used a simple approach to estimate the internal dose due to the administration of radiopharmaceutical with an expression as (Mc. Parland B., 2010) :

$$D_{r_T}(t) = \tilde{A}_{r_S}(t)\, S(r_T \leftarrow r_S; t) \qquad (2)$$

Where :

$\tilde{A}r_s$: cumulated activity in the source region that have occurred up to time t.
$S(r_T \leftarrow r_S;t)$: S-factor, a factor which related to the absorbed dose in the target region r_T to the cumulated activity in source region r_S at time t.

The MIRD society has released several publications in MIRD methods, which have been acknowledged as MIRD pamphlets. The pamphlets have assisted the nuclear medicine researchers to estimate the absorbed radiation dose regarding the administration of radiopharmaceutical in nuclear medicine procedures. A series of MIRD pamphlets have been listed in the Table 1 below (MIRD Publications, 2013).

The MIRD method has been referred for many studies in internal doses assessment, and finally it becomes a standard method in internal dosimetry assessment. Other methods which are available are the method which is established by the International Commission on Radiation Protection (ICRP) and the Radiation Dose Assessment Resource (RADAR) website, an internet based internal dose assessment website (Stabin M, 2006).

ICRP published many reports related to the estimation of internal dose for many radiopharmaceuticals. The first publication in dose to patients from radiopharmaceutical was published in 1971 as Publication 17, and continued by publication 53, 62 and 106.

Table 1. A series of Medical Internal Radiation Dose pamphlets.

No.	TOPICS
1	Discussion of MIRD internal dose technique
3	Photon absorbed dose for small objects
5	Estimates of Absorbed Fractions for Monoenergetic Photon Sources Uniformly Distributed in Various Organs of a Heterogeneous Phantom
11	Absorbed Dose per Unit Cumulated Activity for Selected Radionuclides and Organs
12	Kinetic Models for Absorbed Dose Calculation
13	Specific Absorbed Fractions for Photon Sources Uniformly Distributed in the Heart Chambers and Heart Wall of Heterogeneous Phantom
14	A Dynamic Urinary Bladder Model for Radiation Dose Calculations
15	Radio nuclides Values in a Revised Dosimetry Model of the Adult Head and Brain
16	Techniques for Quantitative Radiopharmaceutical Biodistribution
	Data Acquisition and Analysis for Use in Human Radiation Dose Estimates
17	A Dynamic Urinary Bladder Model for Radiation Dose Calculations
18	Administered Cumulated Activity for Ventilation Studies
19	Radiation Absorbed Dose Estimates from 18F-FDG
20	Radiation Absorbed-Dose Estimates for 111In- and 90Y-Ibritumomab Tiuxetan
21	A Generalized Schema for Radiopharmaceutical Dosimetry—Standardization of Nomenclature
22	Radiobiology and Dosimetry of Alpha-Particle Emitters for Targeted Radionuclide Therapy
23	Quantitative SPECT for patient-specific 3-dimensional dosimetry in internal radionuclide therapy

Those publications provide the estimation of absorbed dose, biokinetic models, and effective doses for more than a hundred radipharmaceuticals to assist the nuclear medicine practician in protecting the patients in nuclear medicine procedures (ICRP, 2008).

MIRDOSE computer code

In order to facilitate the calculation of organ absorbed doses following an injection of radiopharmaceutical, a Windows-based computer code called MIRDOSE was developed by the United States Society of Nuclear Medicine. This program uses lookup tables of S values and requires the user to enter the name of radiopharmaceutical, the preferred phantom, and the residence time τ for each source organ, which is defined as:

$$\tau_h = \frac{\tilde{A}_h}{A_0} \qquad (3)$$

Where

A_0 : the activity administered at time zero

$\tilde{A}_h$: the total number of disintegrations from the radionuclide located in the source

The first MIRDOSE was released in 1987, and updated by MIRDOSE 2 and MIRDOSE3. The software allows the users to provide inputs such as the name of the radiopharmaceuticals, the phantom models which want to be used, the residence time for the radiopharmaceutical. The result of MIRDOSE calculation are radiation dose per unit administered activity (Siegel et al., 1999). A front page of MIRDOSE3 is shown in Figure 1.

Quantification Method for Internal Dosimetry Assessment

A planar imaging from diagnostic procedure using gamma camera or scintillation camera can be utilized to quantify the image to find the uptake activity in the organs. A method called conjugate view method has been referred in the MIRD Pamphlets No.16, the Techniques for Quantitative Radiopharmaceutical Biodistribution Data Acquisition and Analysis for Use in Human Radiation Dose Estimates (Siegel et al., 1999). The method has been acknowledged by few studies (Shahbazi-gahrouei, 2012; Pereira et al., 2010; Buijs et al., 1998), by employing the number of count on the images of chosen organs, and in the posterior and anterior position. It follows the equation:

$$A = \sqrt{\frac{C_A C_P}{e^{\mu T}}} \; x \; \frac{1}{C} \qquad (4)$$

C_A : counts in the region of interest in anterior images

C_P : counts in the region of interest in posterior images

μ : linear attenuation coefficient of 99m-Tc (cm^{-1})

C : Calibration factor of the gamma camera (counts/Bq.s)

T : body's thickness (cm)

The method has been beneficial to be applied in imaging quantification using physical phantoms and patient images (Shahbazi-gahrouei et al., 2012), and a correction method also has been applied by reducing the activity of the background (Pereira et al., 2010; Buijs et al., 1998). Furthermore, in order to find the organ dose, a dose factor (S value) for the organs will be multiplied by the activity of the radiopharmaceutical in the organ (Buijs et al., 1998) :

$$D = A \times S \qquad (5)$$

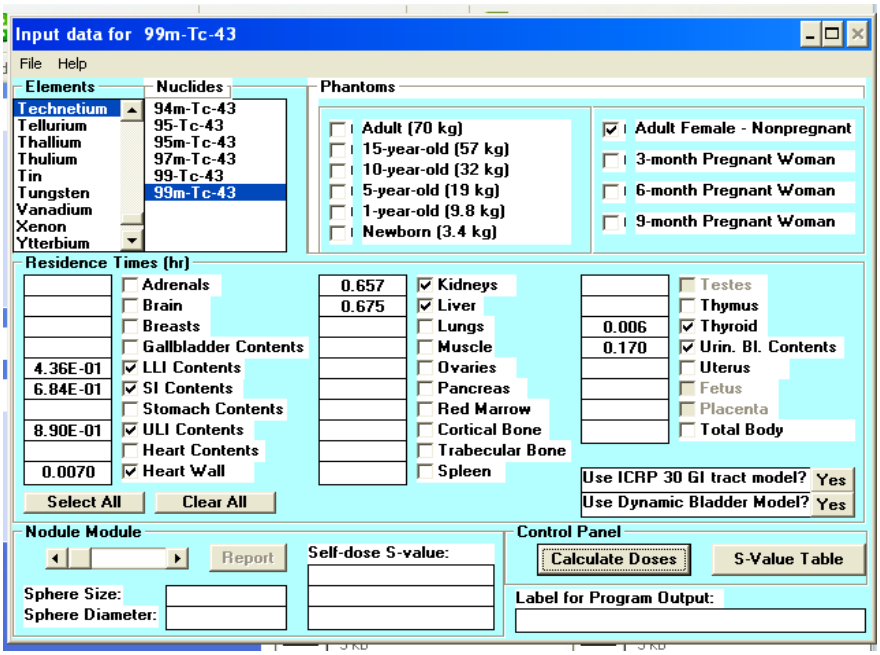


Figure 1. The front page of internal dosimetry tool of MIRDOSE3

Materials and Methods

The study was performed on 20 adult female volunteers, with range of age 20-66 year, after the administration of 99m-Tc Sestamibi at about 370 MBq (10 mCi). Patients were scanned with the dual-head gamma camera AnyScan MEDISO at 45-60 minutes after the injection. In order to perform the image quantification, a conjugate view method was employed on total body scan images. At about twelve regions of interests (ROI) with a similar size was drawn for each image of patient, consists of six ROIs from anterior position and posterior position, for thyroid gland, heart, liver, right and left kidneys, and bladder. Finally the counts from all the ROIs have been subtracted with the background value and been calculated with the equation (3). By using equation (3), where C_A and C_P are the counts in ROI anterior and posterior, respectively, the linear attenuation coefficient of 99m-Tc (μ, in cm^{-1}) was investigated by placing a source of 99m-Tc (in syringe) with one of installed detectors. The coefficient was obtained by plotting a series of data from the number of count which is obtained by placing few slices of acrylic between the source and camera (Pereira et al., 2010). A calibration factor also was determined by counting a small source of 99m-Tc (in the small syringe) for few minutes without any acrylic slab, in order to get the factor in counts/Bq.s.

Results and Discussion

In general, there are two significant findings from the imaging quantification of target organs. The uptake process has been found in different pattern for each patient. Meanwhile, the average figures can be seen by combining all the result and presenting in Figure 1. From the graph in Figure 1, it can be seen that patient no. 2 and 13 have particular uptakes of Sestamibi in the bladder, while the rest of results show that most of patients have normal uptake in their organs. The results of image acquisition for patient no.2 and 13 are presented in Figure 2. This condition might be caused by the patients had not emptying the content of their bladder prior to the acquisition. Hence, for future investigations, a standard procedure must be applied, that is the patients should have urinated after the injection. Obviously, a high uptake in one organ can interfere the acquisition of other surrounding organs due to either high background or overlapping images. Another significant feature also exists in the uptake in the liver of similar patients (patient 2 and 13). Hence, a further investigation has been recommended by the Nuclear Medicine specialists.

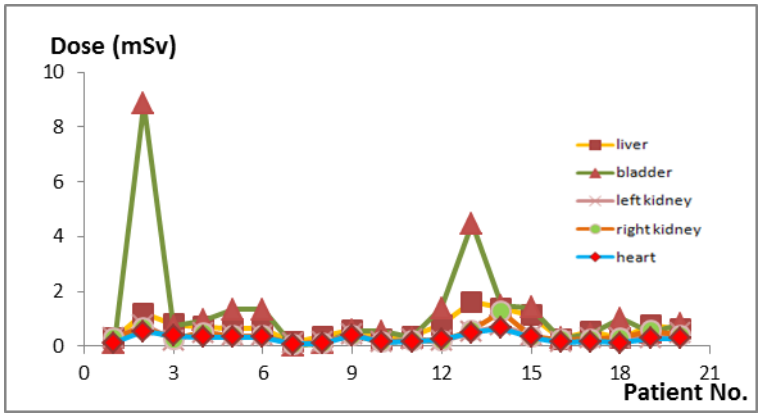
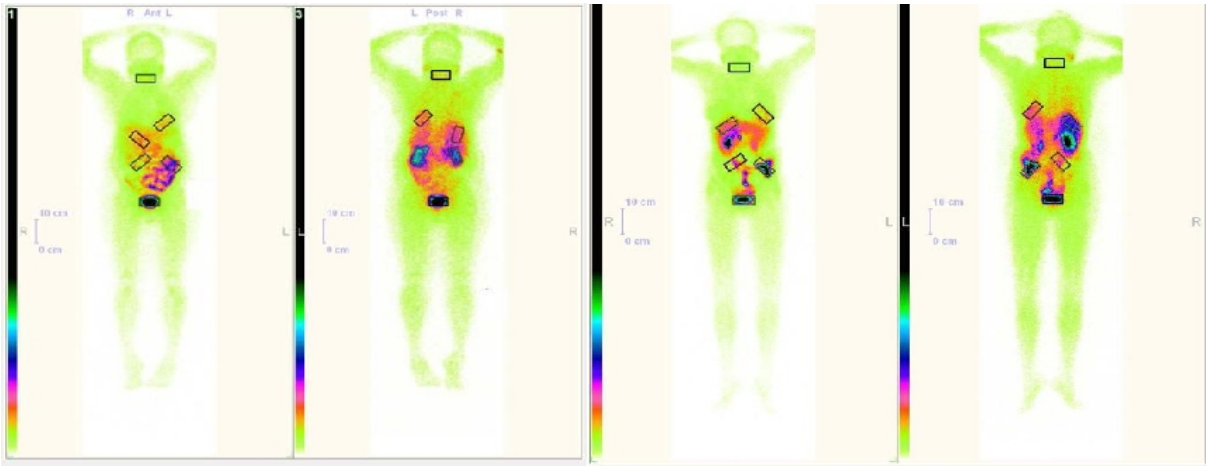


Figure 1. The result of quantified dose among 20 patients



(a) (b)

Figure 2. Acquisition image of patient no.2 (a) and patient no.13 (b).

By combining all the results, we can see that the quantification of imaging can be used as a representation of how the abnormality procedures in nuclear medicine can be analyzed in quantitatively, and not merely by looking up the result of image acquisition (qualitatively). In the next studies, it is better to set the range of normal uptake values by analyzing the image in two methods, quantitavely and qualitatively.

In addition, a comparison between quantified dose from image acquisition and MIRDOSE3 calculation, has been done by entering the input data such as, the residence time for each organ, the name of radiopharmaceutical, and the adult-female phantom. As a result, a value of absorbed dose per Mega Becquerel was obtained in mSv/MBq. Thus, to find the absorbed dose, a multiplication with the amount of administered dose should be applied. From Figure 3, it can be seen that patient no. 2 and no.13 have higher uptake for the bladder and liver, while other patients, for example patient 9 and 15 in Figure 4, have almost similar pattern where the dose to the bladder and the liver are less than 1 mSv.

A previous study has been performed with a similar comparison between quantification of imaging versus MIRDOSE3 , resulting in a good agreement between quantification of imaging and MIRDOSE3 calculation (Stabin M, 1996). It also has been reported that there are several factors that influence the result of image quantification, namely body thickness, device sensitivity, and effect of overlapping tissue, which might be resulted in unreliable quantification. In that previous study, a range of time were applied, so there are few observation which be done few times in a specific time such as t 10, 60, 90, 180 minutes after the injection. Hence, for further studies, it would be better if it could be noted precisely that the time between the administration and the acquisition of image, in order to make a correction about the timing, since 99m-Tc has a really short physical half life (about 6 hours). The confirmation about time correction should be performed to evaluate whether the cause of the different result between the quantification of imaging and the calculated dose from MIRDOSE3 is caused by the timing factor or not. In addition, since the study has been done for administration of 99m-Tc Sestamibi, a further investigation is possible to be performed for other diagnostic procedures, such as 99m-Tc MDP for bone scan, 99m-Tc MAA for lung perfusion scan, and other radiopharmaceuticals.

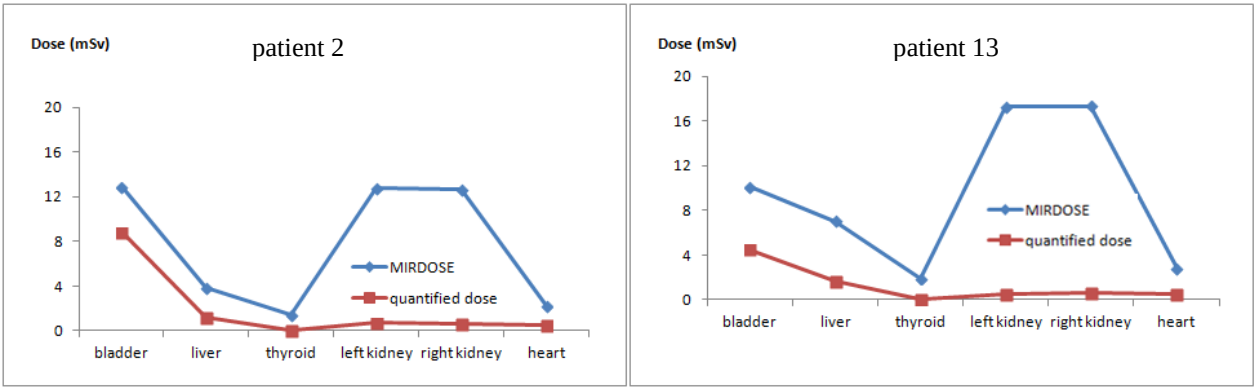


Figure 3. Comparison of quantified dose from image acquisition and calculated dose from MIRDOSE3 in patients no. 2 and 13.

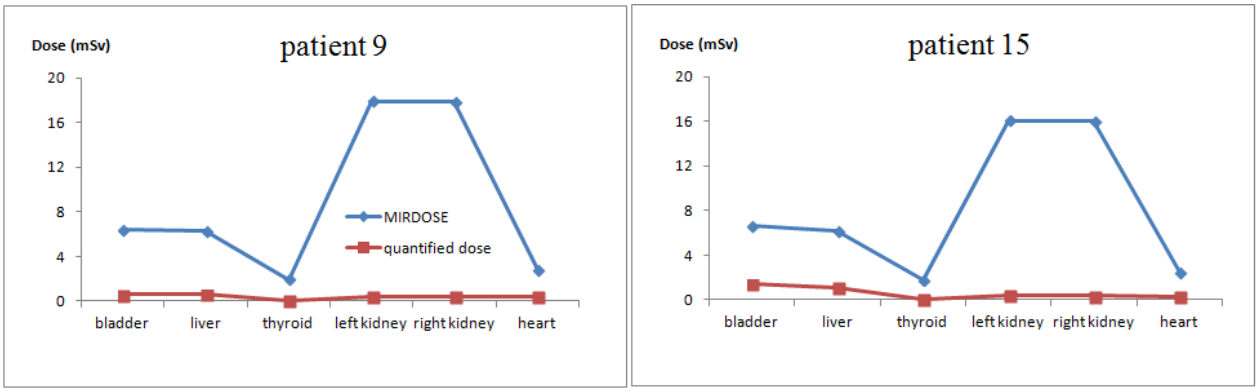


Figure 4. Comparison of quantified dose from image acquisition and calculated dose from MIRDOSE3 for patients no. 9 and 15.

Conclusion

As a result, the image quantification in 20 patients enrolled in the study was mostly did not consistent with the result from MIRDOSE3 calculation. Every patient have her own specific dose distribution where it might reflect the variation of anatomical and physiological condition of each patient. It also shows that in further investigation, another correction method should be applied, and a time based investigation would be applied.

Acknowledgements

We thank Prof. Richard Smart from the University of New South Wales for assistance with the presentation materials in internal dosimetry and suggestions. The work was supported by all the staff of Department of Medicine and Nuclear Technology, Center for Metrology and Radiation Safety Technology, National Nuclear Energy Agency of Indonesia (BATAN)

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DISCUSSION

dr. B. Okky Kadharusman, PTKMR BATAN:

Question : What is the benefit of internal dosimetry assessment for patients.

Answer : The internal dosimetry assessment for patients will be very useful for patients especially in the case of radionuclide therapy will be delivered, where there is an organ which has been considered as organ at risks. For example, in the case of Peptide Receptor Radionuclide Therapy, kidney has been considered as an organ at risk. The assessment will also give a prediction how much the dose will be delivered in the therapy when a patient specific dosimetry has been conducted prior to the treatment.

Expression of Major Vault Protein in Cervical Cancer Before Treated with Chemoradiotherapy *)

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Abstract. **Cervical cancer is the most often case found in Indonesia. Chemoradioherapy is the primary treatment of this cancer at the locally advanced stage that given by concurrent radiotherapy with chemotherapy in the same time. Response of chemoradiotherapy is influenced by biological and physical factors. Major Vault Protein (MVP) is ribonucleoprotein contributes to drug resistance in some cancers. The purpose of the research was to determine the relationship between the expression of MVP and p53 LI AgNOR value, MIB-1 LI before treatment and association between MVP with chemoradiotherapy clinical response of cervical cancer. Twenty-one microscopic slides taken from cervical cancer tissue biopsy patients before treatment were stained with MVP, p53 and MIB-1 by immunohistochemistry technique and AgNORs staining. After the completion of treatment, chemoradiotherapy clinical response was observed by pelvic control method. As the result, before chemoradiotherapy there was a relationship between MVP and AgNOR value (p=0,05) and no relationship between MVP with p53 index (P=0.72) and with MIB-1 LI (p=0.63). After chemoradiotherapy there was no association between MVP expression with chemoradiotherapy response. In conclusion, there was relationship between MVP expression with AgNORs value, no related to p53 LI and MIB-1 LI before treatment and no association between expression of MVP with chemoradiotherapy clinical response.**

Keywords: Major Vault Protein, p53, MIB-1, AgNOR, chemoradiotherapy, Cervical Cancer

*) The complete paper will be published in Atom Indonesia Journal.

Calibration of Non Invasive X-ray Tube Potential Analyzer Using Spectroscopy System

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Abstract. **Non invasive beam analyser is one of popular compliance test tool to check X-ray generator and tube performance. Tube potential is one of important parameter that should be checked on each compliance test, therefore the need of calibration procedure to calibrate X-ray tube potential is become necessary. The experiment was done at Secondary Standard Laboratory Jakarta using X123CdTe spectroscopy system and YXLON TU230-D03 X-ray tube with constant potential generator. The spectroscopy calibration was done using Am-241 and Ba-133, and the calibration procedure was done without collimator. X-ray spectrum measurements were done with collimator and without collimator. The purpose of this measurement was to compare beam collimation and absorber insertion techniques to reduce X-ray beam intensity. For collimated X-ray beam measurement the filtration was arranged to make the 1st HVL is following IEC61267 guideline with collimator size of 400□m. Uncollimated beam measurements were done by increasing the Cu absorber between X-ray source. The spectroscopy detector was placed at 100 cm away from the source, and the additional filter or absorber was placed at 50 m away from the X-ray source. Two non invasive kVp meters which were Unfors R/F and Fluke 4000M+ were calibrated using the spectrum tail ends as the real kV peak value. The difference between X-ray tube generator panel was less than 1 keV. The result showed the spectrum tail ends was sufficient to choose as tube kVpeak value. Non-invasive calibration using spectrum tail ends showed small differences which was less than 1.02kV and 1.24kV for Unfors Xi and 4000M+. The spectrum tail from absorber technique was sharper than the collimated beam. This was very useful since it will remove ambiguity in choosing the last point of the spectrum tail. Reducing absorber to reduce intensity was giving the best result to calibrate non invasive kVp meter.**

Keyword - Spectrum, Spectroscopy, Calibration, Non invasive, Beam analyser

Introduction

The trend of diagnostic x-ray imaging is increasing in Indonesia, and every X-ray equipment in medical application should passed the compliance test. One of popular compliance test tool to check x-ray generator and tube performance is Non Invasive Beam Analyser. Tube potential is the most important parameter that must be checked, therefore the need of calibration technique to calibrate x-ray tube potential is very important. X-ray tube peak voltage can be measured using invasive technique by applying high voltage divider that connected to anode and cathode with high tension cable (International Electrotechnical Commission, 2002).

Other method to measure the tube peak voltage is using spectroscopy system. The former method is the most direct approach to measure tube peak voltage using its electrical quantity, this approach can measure the presence of ripple in the electrical wave form. Spectroscopy measurement offers another approach to measure the peak voltage. This method measures the peak voltage from X-ray spectrum due to its physical interaction and this method does not measured the electrical ripple. Some papers present the ability of spectroscopy system to measured peak voltage (Oliviera et al, Dalla et. Al. 2001, Potiens et. al., Terini et. al. 2004). In order to improve accuracy in detecting the peak voltage, high resolution spectroscopy system is required (Terini, 2004).

Diagnostic X-ray gives very high dose intensity, and this is a problematic issue in spectroscopy measurement. High dose intensity will reduce spectroscopy counting rate, increase dead time and at the worst situation it will create electrical noise in the electrical system. There are three common solutions to reduce dose intensity which are using a small collimator, increase measurement distance or adding more absorber. This work will uses collimator and adding more absorber to reduce the dose intensity, and to see wether the absorber will increase the spectroscopy measurement sensitivity and improve the spectrum shape. Nowadays, compact and electric cooling semiconductor spectroscopy is available, this type of spectroscopy system can work in normal room temperature and convenient to use. This present work is our initial study to investigate our measuring system capability to conform ISO17025 guideline.

Materials and Methods

The experiment was done at Secondary Standard Laboratory Jakarta using X123CdTe spectroscopy system and YXLON TU230-D03 x-rays tube with constant potential generator. The spectroscopy calibration was done using Am-241 and Ba-133, and the calibration procedure was done without collimator. Table 1 show the energies used for spectroscopy calibration.

X-ray spectrum measurements were done with collimator and without collimator. The purpose of this measurement is to compare beam collimation and absorber insertion techniques to reduce X-ray beam intensity. Those techniques were done to make linier like spectrum tail, since it is very important to determine the peak voltage. When collimator is used, there is a big probability of X-ray scatter will happen inside the collimator and reach the detector and affected the spectrum tail. The scatter could increase low energy X-ray spectrum and will increase dead time and reduce counting rate since these low energy X-rays tend to increase pulse pile up in detection system. Increasing the absorber between X-ray source and detector can reduce low energy scatter, since the absorber itself will remove low energy spectrum. Disadvantage of this technique is, we need to increase the tube current due to heavy filtration of the X-ray beam. When this technique is used, we do not need to use collimator because the intensity is low enough for spectroscopy detector to measure without loosing its count rate and creating dead time effect on the system. The absorber being use in this work was Cu with various thicknesses to ensure that the spectroscopy system received the adequate intensity without creating dead time and reducing count rate, and for collimator technique the collimators size use was 400 m. The tube current applied is varied depend on the X-ray tube potential. For collimated X-ray beam measurement the filtration is arranged to make the 1st HVL is following IEC61267 guideline (IAEA 2007, IEC 2005). During the measurement the spectroscopy system dead time was kept below 6%. The filtration list is shown in table 2. The spectroscopy detector was placed at 100 cm away from the source, and the additional filter

Table 1. Am-241 and Ba-133 energy spectrum.

Source	Energy (kev)	Probability (%)
Am-241	13.90	42.00
	17.80	19.40
	20.80	4.90
	26.30	2.40
	59.50	35.90
Ba-133	30.97	24.60
	35.20	8.84
	81.00	91.00

Table 2. Filter and tube current setting for collimated and uncollimated beam.

Tube Potential (kV)	Collimated Current (mA)	Collimated Filtration (mmAl)	Uncollimated Current (mA)	Uncollimated Filtration (mmCu)
40	-	-	5	1.99
50	1	2.62	0.5	1.99
60	30	3.1	1	5.98
70	10	3.32	1	5.98
80	3	3.4	0.5	6.99
90	1	3.55	5	14.99
100	0.5	3.62	5	19.99
110	-	-	1	19.99
120	1	4	0.5	19.99
130	-	-	0.5	19.99
140	-	-	0.5	19.99
150	1	5	0.5	19.99

or absorber was placed at 50 cm away from the X-ray source.

The X-ray spectrum peak measured at the end of the xray spectrum tail, and it compared with non invasive kvp meters. Two non invasive kVp meter used in this work, which are Unfors R/F and Fluke 4000M+. On all measurements all detectors were placed at 100 cm distance from the source, and since the X-ray generator used in this experiment is constant potential, the expected spectrum tail should be linear and the tube voltage peak is determined by the end of the spectrum tail. The tube potential parameter measured by non invasive kVp meter in this work is kV average, because the X-ray generator is constant potential the kV average will equal to kVpeak.

Results and Discussion

Figure 1 is showing Am-241 and Ba-133 spectrum, and spectrum peak captured by the spectroscopy system is used to create the calibration curve. Relation between channel and energy during calibration process showed regression correlation near to 1, and the calibration function is Energy = 0.171 * Channel -0.206 with R^2=1.00 as shown in figure 2.

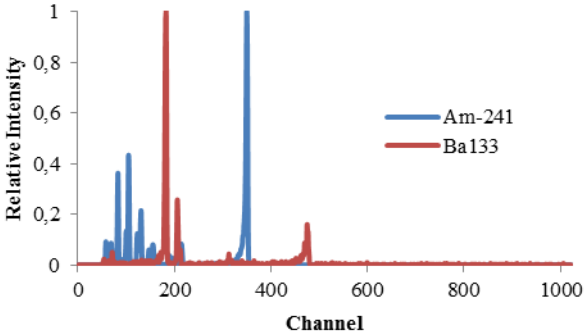


Figure 1. Am-241 and Ba-133 Spectrum used for calibration.

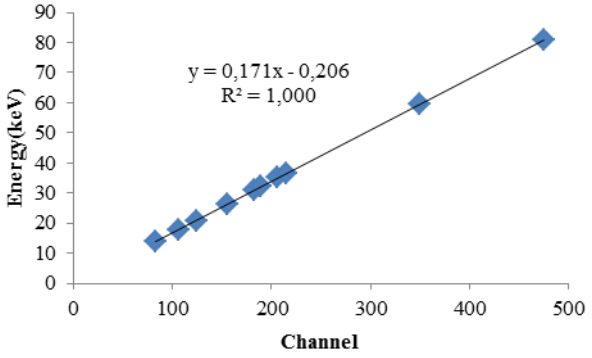


Figure 2. Calibration curve of CdTe Spectroscopy system.

All measured spectrum is not corrected for electron escape, since this study is focus on spectrum tail. Spectrum for uncollimated X-ray spectrum is shown in Figure 3. At energy below 30keV there are high intensity peak, it happened because electron escape peak event at CdTe (Redus, 2008), and it does not happened for energy above 40keV. When Cu absorber is applied, the low energy part is reduced and the spectrum is shifted to higher energy level due too heavy filtration of the X-rays spectrum. Complete result of spectrum measurements using Cu absorber is shown in figure 4. The electron escape peak effect is reduced at tube potential above 40kV, therefore more precaution is required for low kV measurement.

From all measurements, the spectrum tail of uncollimated beam is better than collimated one. The collimator uses during measurement is 9 cm long, and if the collimator is not well align with the X-ray source, there will a possibility that the primary beam scattered before it reach the detector. This effect will make the spectrum tail does not sharp enough when compare with uncollimated beam, this happened due too low energy scatter inside the collimator.

Comparison result of collimated and uncollimated beams are shown in figure 5, we saw that the spectrum tail for uncollimated beam is sharper than collimated beam. This spectrum is used as reference value to determine the tube potential peak. The result of tube potential peak is shown in Table 3 and from the spectrum tail end it showed that the accuracy of spectrum tube potential is greatly depend on detector resolution (Terini 2004).

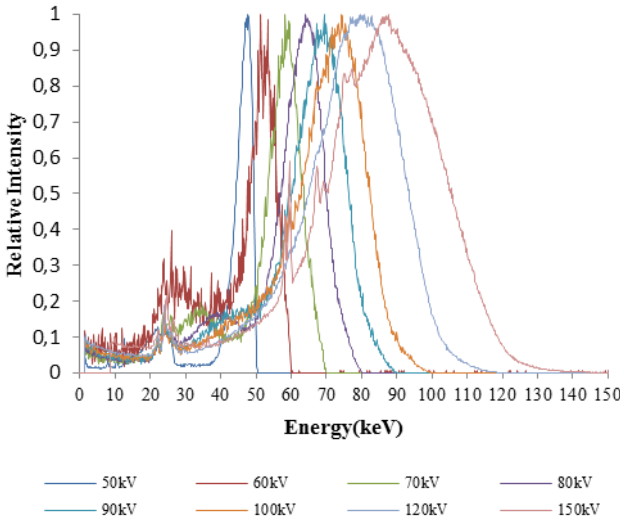


Figure 3. Measurement spectrum for collimated beam with Al Absorber.

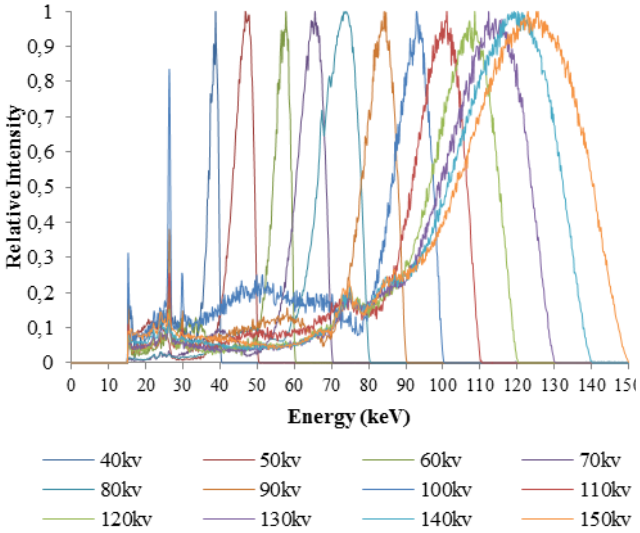
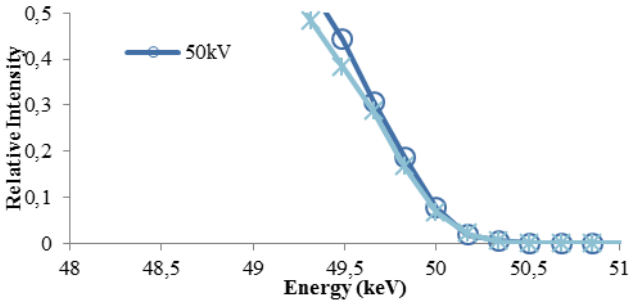
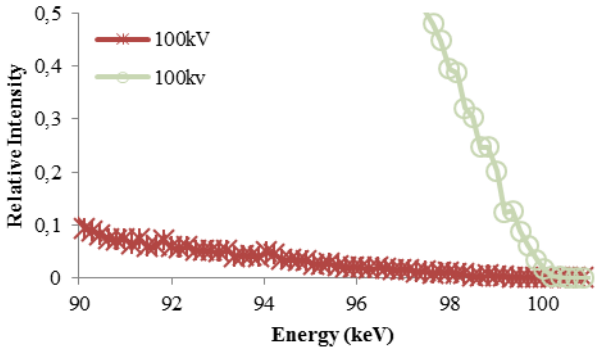


Figure 4. Measured spectrum of X-ray beam using Cu absorber and uncollimated detector.

To ensure stability of the X-ray tube generator and detector reproducibility, measurement using non invasive beam analysers repeated minimum 5 times, and the maximum variation is less than 2.25kV for Unfors Xi and 2.45kV for 4000M+. Measurement mode selected for Unfors Xi and 4000M+ is kV average due too low tube current being used and the intensity is not high enough for unfors Xi to measure in kVp mode, therefore same mode is selected for 4000M+.The reference spectrum used for calibration is spectrum from uncollimated beam and showed in Table 3, since uncollimated beam spectrums has better tail. X-ray tube potential measurement using non invasive kVp meter



(a)



(b)

Figure 5. uncollimated (o) and collimated (x) spectrum tail ends of (a) 50kV, and (b) 150kV tube potential, the collimated spectrum is sharper than uncollimated beam.

were done and compared with spectroscopy measurement, and the result was shown in Table 4. All measurement showed good consistency and maximum deviation against spectroscopy system between Unfors Xi and 4000M+ is 1.21% and 2.45%. These data show that all the spectroscopy system is able to use as calibration tool to calibrate non invasive kVp meter.

Table 3. spectrum tail end of uncollimated beam.

Tube Voltage (kV)	Spectroscopy spectrum peak (keV)	Δ (kev)
40	40.27	0.27
50	50.51	0.51
60	60.42	0.42
70	70.66	0.66
80	80.91	0.91
90	90.30	0.30
100	100.38	0.38
110	110.62	0.62
120	120.53	0.53
130	130.43	0.43
140	140.51	0.51
150	150.58	0.58

Table 4. Calibration result of Unfors Xi and 4000M+.

Spectroscopy peak (keV)	Unfors Xi (kV average)	Δ (%)	4000M+ (kV average)	Δ (%)
40.27	-	-	39.99+0.34	0.70
50.51	49.90+0.12	1.21	49.27+0.20	2.45
60.42	60.20+0.06	0.36	59.52+0.14	1.48
70.66	70.50+0.18	0.23	69.78+0.29	1.24
80.91	81.00+0.33	0.11	80.20+0.5	0.88
90.3	91.30+0.24	1.11	90.54+0.53	0.26
100.38	101.40+0.49	1.02	101.00+0.39	0.62
120.53	121.20+0.51	0.56	-	-
150.58	150.80+2.25	0.15	-	-

Conclusions

Application of collimator during measurement will increase probability of scatter radiation created from bounced X-ray inside the collimator that will reach spectroscopy detector. Spectrum modification using Cu absorber gave big advantage in finding the tail end of X-ray spectrum. The spectrum tail from absorber technique is sharper than the collimated beam. This is very useful since it will remove ambiguity in choosing the last point of the spectrum tail. Increasing absorber to reduce intensity gave the best result to calibrate non invasive kVp meter.

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Calibration of High Dose Rate of Ir-192 Brachytherapy Sources

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Abstract. **Brachytherapy is one of modalities in the cancer treatment that has been developed. Ir-192 radiation source is the one that has been used in brachytherapy, due to the high dose rate it produces so that the time needed to expose patient is only in the order of minute. To be accurately use with low uncertainties, Ir-192 source should be calibrated and its dosimetric characteristics should also be known. Calibration of the air kerma rate of Ir-192 was carried out by using a Farmer type ionization chamber, while dosimetric characteristics were determined using gafchromic film. The calibrated brachytherapy sources were six of those located in six different hospitals in Indonesia. The results show that the method of measurement carried out in this study is consistent with the measurement performed by Ir-192 brachytherapy source manufactures, whereas the dosimetric characteristics determination shows that the isodose obtained were within the tolerance limit. In addition, the good result of calibration also shows that the SSDL Jakarta is ready to give a significant contribution to support the hospitals to perform calibration of their Ir-192 brachytherapy sources.**

Keyword: *Brachytherapy, calibration, Ir-192, dosimetric characteristic.*

Introduction

Brachytherapy is one of modalities in the cancer treatment that has been developed. In clinical applications, brachytherapy using different methods is compared to teletherapy. Brachytherapy is done by putting the source at a position very close to the cancer cells to be killed. Seen from a technical perspective, brachytherapy source used is very different from the teletherapy source. Brachytherapy using a source with a relatively small activity compared to teletherapy. Size of brachytherapy sources about 1-17 mm while teletherapy using Co-60 source size can reach 20 mm. Very small size is a challenge faced by medical physicists in determining the dose rate brachytherapy source.

Ir-192 is a source that is widely used in brachytherapy. The high dose rate (generally greater than 12 Gy/h) causes the source widely used because the time needed to expose patients is only in the order of minutes. However, a short half life (T ½ = 73.83 day) (Podgorsak et al., 1993) causes the hospital to replace this source in the span of about four months.

The manufacturer of the Ir-192 sources generally issues a calibration certificate of source strength with an overall uncertainty of ± 5%. Considerable uncertainty makes the user must perform an independent calibration before the source is used for clinical purposes.

The recommended quantity for specifying the strength of brachytherapy source is an air kerma strength. Air kerma strength is defined as the product of air kerma rate at calibration distance, *d*, in the open space, *K(d)*, measured along the source transverse bisector, and the square of the distance, *d*. Air kerma strength is one of component dosimetric characteristics beside geometry factor, radial dose distribution and anisotropic function that should be taken into account for clinical purposes (Rivard et.al., 2004).

There are two methods of calibration Ir-192 source, using the well type ionization chamber and the Farmer type ionization chamber in air (Baltas et al., 1999). The well type ionization chamber method is recommended for clinical environments to achieve greater precision because of its geometry can be repeated (reproducible). However, this method has a difficulty because it is not easy to obtain calibration factors directly due to the photon energy spectrum of Ir-192 is very wide.

The calibration factor of Ir-192 gamma ray beam was first derived from the average of the calibration factors for the 250 kV X-rays and the Co-60 gamma ray beam (Ezzell, 1989). This method was later modified by defining the calibration factor of air kerma Ir-192 as the average calibration factor for the 250 kV X-rays and Cs-137 gamma ray beam (Goetsch et al., 1991). For both methods, the calibration factors for all energies were obtained with 0.321 $g.cm^{-2}$ thick buildup cap. International Atomic Energy Agency (IAEA, 2002) and (Nath et al., 1991) recommend the use of ionization chamber with air kerma calibration factor obtained by weighted interpolation between 250 kV X-rays and Co-60 or Cs-137 gamma rays energies.

The German standards have recommended the use of ionization chamber with a volume greater than 1.0 cc for in-air calibration (Deutsches Institut fuer Normung, 1993). The multiple-distance measurement method was adopted for in-air calibration to make a correction for an error in positioning of the chamber and scattered radiation. However, this method turned out to be very time-consuming and unnecessary to use for routine calibration. Therefore, it is suggested that the results of in-air calibration should be used for deriving a calibration factor for an alternative, quicker technique using simpler instrument.

Organization of medical physics in Germany (Deutsche Gesellschaft fuer Medizinische Physik, 1989) recommends the use of specific solids to obtain a higher reproducibility and accuracy in the calibration of brachytherapy sources. This method resolved the problem due to not only positioning error and room scatter but also contributes an uncertainty since the calibration factors derived from the measurements in solid phantom with calibrated references source. The ESTRO guidelines (ESTRO, 2004) recommends to using some calibration jig with source position at 10 cm distance from 0.6 cc ionization chamber to reduce placement uncertainty and scattering contribution.

In Indonesia, Ir-192 brachytherapy source has been quite widely used in several hospitals. Unfortunately, Ir-192 source was never calibrated by almost all hospitals. The dose rate used for treatment patients is the result of Treatment Planning System (TPS) Algorithm based on decay table that have been input by distributor of brachytherapy machine.

In this study, calibration or determination of the air kerma rate Ir-192 brachytherapy sources performed using a Farmer type ionization chamber placed on a special calibration jig (see Figure 1). It is also to determine the dosimetric characteristics of Ir-192 source using a gafchromic film dosimeter. The purpose of this study is to get the method to determine the dosimetric characteristics and air kerma rate of Ir-192 brachytherapy source is simply to do for practical purposes the hospital routine.

Materials and Methods

Determination of Ir-192 air kerma rate

Measurement of air kerma rate of Ir-192 brachytherapy sources was performed using 0,6 cc ionization chamber NE 2571 coupled with a Keithley 6487 electrometer. Based on IAEA recommendation [6], the detectors was calibrated beforehand using Co-60 and X-ray 250 kV beams. Calibration was performed at the Secondary Standard Dosimetry Laboratory (SSDL) BATAN Jakarta which has been designated to be the

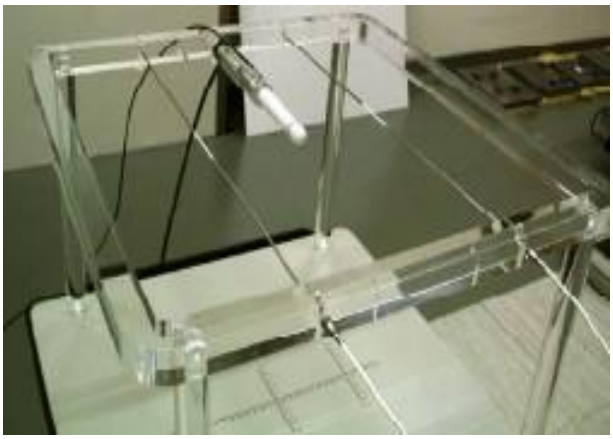

Figure 1. Special calibration jig for Nucletron mHDR machine.

National Dosimetry Laboratory (PerKa BAPETEN, 2006).

Measurements of air kerma rate were performed for brachytherapy Microselectron HDR V2 Nucletron machines and Gammamed 12i HDR /PDR machine as well. In each measurement, the ionization chamber was placed at the calibration jig.

Air kerma rate, K_R, was determined at a reference distance, d_{ref} of 1 m, and calculated by the equation (IAEA, 2002):

$$K_R = N_K \left(\frac{M_U}{T}\right) . k_{air} . k_{sca} . k_n \left(\frac{d}{d_{ref}}\right)^2 \quad (1)$$

Where N_K was air kerma calibration factor and *Mu* was measurement reading during T measurement time, while k_{air} , k_{sca} , and k_n was Air attenuation, Scattering, and non-uniformity corection factor.

Determination of Ir-192 air kerma calibration factors, $N_{K,Ir}$ performed using 250 kV X-ray source and Co-60 based on the equation:

$$N_{K,Ir} = 0{,}8\, N_{K,250kV} + 0{,}2\, N_{K,Co} \quad (2)$$

Where $N_{K,250kV}$ was air kerma calibration factor performed using 250 kV X-ray beam and $N_{K,Co}$ was air kerma calibration factor performed using Co-60 beam

Determination of dosimetric parameter of Ir-192

Dosimetric parameter was determined by dose distribution of Ir-192 source. Dose distribution measurement was performed using gafchromic EBT2 film detector which was read using Microtek TMA1000XL scanner. The data obtained were

evaluated using FilmQA software by 3cogntionLLC, New York.

Before the measurement of dose distributions HDR Ir-192 source was done, the film gafchromic was calibrated beforehand by exposing the film with various dose until it reached the saturated area. Based on the darkness level curve or film optic density against dose, the correction factor and film response can be determined. Furthermore, the correction factor was used to correct the darkness level in order to obtain the true value of the dose.

The measurement of dose distribution was conducted by exposing the film which was placed in the air. As a result of that radiation exposure occured blackening on the film, and based on the darkness level that was captured on film, the geometry factor and the anisotropy function of the source could be determined. Based on the obtained dosimetric parameter could be known dose distribution of Ir-192 brachytherapy source.

Results and Discussion

Ionization chamber calibration with 250 kV X-ray beam and Co-60 beams generated calibration factor value $N_{K,250kV}$ and $N_{K,Co}$ amounted to 41,16 mGy/ nC and 41,2 mGy/ nC, so using equation (2) obtained calibration factor of Air Kerma for Ir-192 amounted to 41.168 mGy/ nC. Kerma calibration factor value are then used to calculate the air kerma rate using equation (1).

Table 1 shows the air kerma rate value of Ir-192 brachytherapy source measured by ionization chamber in air for several hospitals and the value of the air kerma rate given by the console of each brachytherapy machine.

Table 1. The result of air kerma rate measurement (cGy/h m^2).

Hospital	Measurement	Console Data	Deviation (%)
Persahabatan, Jakarta	1.854	1.860	0.32
Hasan Sadikin, Bandung	1.670	1.683	0.78
Adam Malik, Medan	1.533	1.502	2.02
Sardjito, Yogyakarta	1.128	1.138	0.89
Sutomo, Surabaya	2.329	2.493	7.04
Karyadi, Semarang	2.143	2.095	2.24

Deviation = [(measurement – calculation)/calculation] x 100%

As shown in Table 1, the deviation between the results of measurements and the data provided by the console is generally quite small, or less than 3.5% as the maximum acceptable deviation in radiotherapy. These results indicate that the method of measurement performed reasonably consistent with measurements conducted by Nucletron and Gammamed as a vendor of Ir-192 brachytherapy source.

However, the deviation of the measurements obtained in Sutomo Hospital Surabaya is quite large. The cause of the difference presumably occurred is due to the use of different calculation time standards. Console brachytherapy machine still uses the standard European time (GMT+2) in the calculation of the air kerma rate, while the application is using a standard western Indonesia time (GMT+8).

The measurement result in Table 1 also indicate that the value of the calibration factor given by the SSDL Jakarta is relatively good. Calibration factor used by the detector in the measurement is the calibration factor obtained from cross calibration procedure performed by the SSDL Jakarta between the standard detector and the field detector. These results also show that cross-calibration results obtained relatively good, as seen from the small difference between the results of measurements and the data console given.

Figure 2 shows the example of irradiation gafchromic films result with Ir-192 brachytherapy source (in this case the results of irradiation in Sardjito Hospital, Yogyakarta). The results are evaluated and analyzed in order to generate the dosimetric characteristic in the term of isodose curves form as given in Figure 3.

The isodose determination is necessary considering brachytherapy sources placed very close with irradiation target in application. Thus, dosimetric characteristics are required to accurately determine the amount of radiation dose given to patients. Figure 3 shows that isodose for all brachytherapy machines used are within the acceptable tolerance of ± 5%.

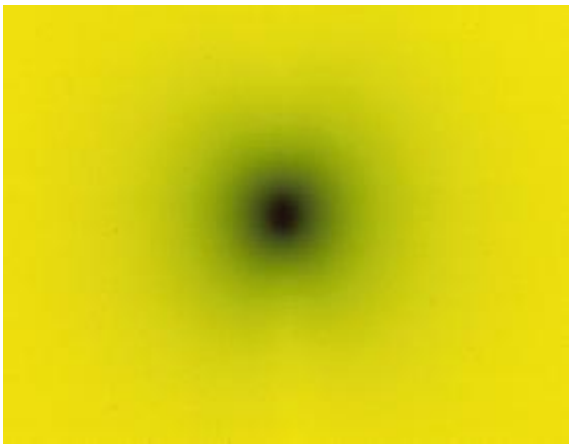

Figure 2. The example of irradiation gafchromic films with Ir-192 brachytherapy source.

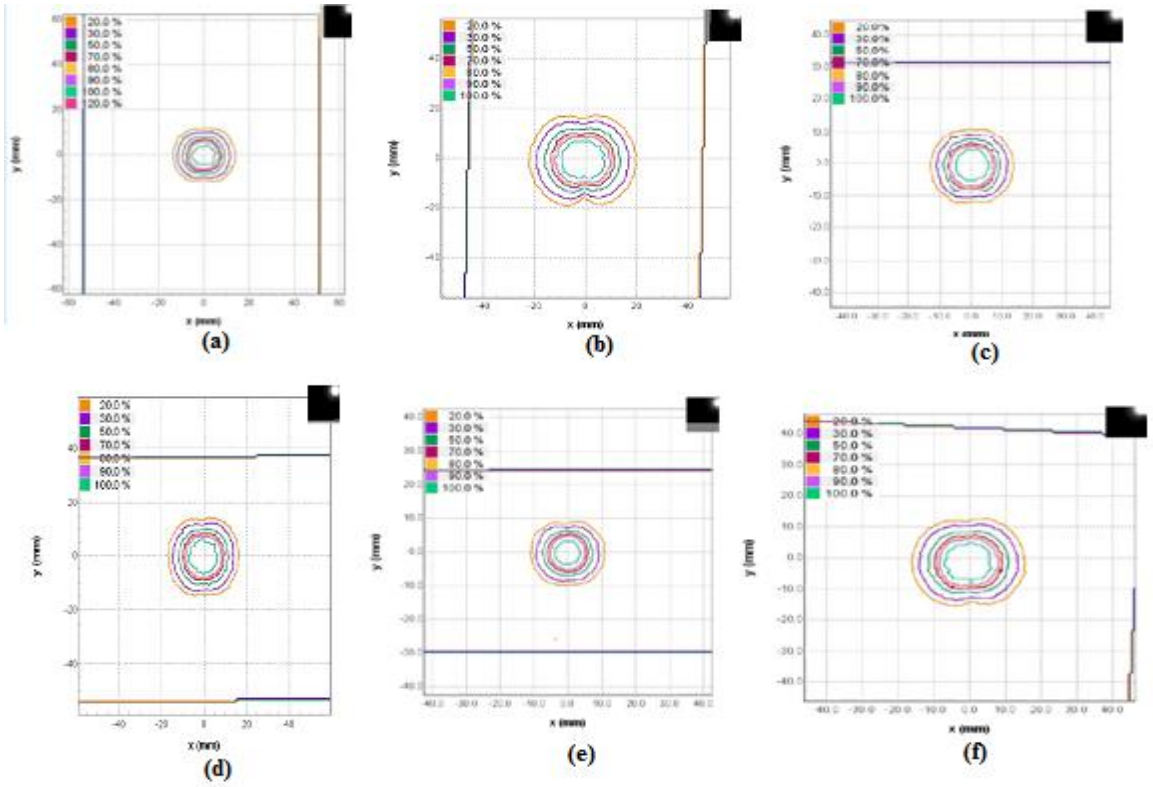

Figure 3. Ir-192 Source isodose curves a) Persahabatan Hospital - Jakarta, b) Hasan Sadikin Hospital – Bandung , c) Sardjito Hospital – Yogyakarta, d) Karyadi Hospital – Semarang, e) Soetomo Hospital – Surabaya, and f) Adam Malik Hospital – Medan.

Conclusions

Based on the results obtained from this study it can be concluded that the method of measurement performed is reasonably consistent with measurements made by Nucletron and Gammamed as a manufacturer of Ir-192 brachytherapy sources, while the determination of the characteristics of the dosimeter shows isodose obtained was within the acceptable tolerance. In addition, a good calibration results shows that the SSDL Jakarta is ready to contribute significantly in supporting the hospital to perform the calibration of their Ir-192 brachytherapy sources.

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DISCUSSION

Dadong Iskandar - PTKMR BATAN :

Question : Why there was any deviation in data between hospitals in your research?

Answer : The deviation is caused by the difference of the scatter correction factor (Ksca) which obtained from the difference of room size between hospitals. Note that in calculation we assume that the scatter correction factor for all hospital = 1.

Comparison of Thorax Patient Dose Using Calculation With CALDose_X 5.0 Software and Direct Measurement With TLD-100

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Abstract. **The use of X-ray as a means to diagnose an illness in human is already wide. To ensure and to monitor the dose received by patient who undergoing radiodiagnostic examination, it required an in vivo dosimeter such as thermoluminiscence dosimeter (TLD) to estimate the entrance surface dose (ESD). Two patient populations were measured their thorax photo and then estimated their dosage by using CALDose_X 5.0 program. The ESD was measured using TLD that was attached in the center of the field of radiation. After the irradiation, the TLD was read using a TLD reader. Estimated dose calculations were performed using the CALDose_X 5.0 software. Parameters included in the calculation were kV, mAs, FFD, output (mGy/mA.s) from the X-rays machine used. Comparative result showed a quite good correlationship between the value of ESD from measurement and that of ESD from the calculation. The correlation coefficient (r^2) was 0.889 for the first population, and 0.936 for the second population. The CALDose_X 5.0 software can be applied to estimate ESD for patient undergoing radiodiagnostic examination since there is a good correlation between measurement using a TLD and calculation using a software. It can be concluded that CALDose_X 5.0 software was reliable to complement the use of TLD. It need a further research to obtain the dose estimation method which appropriate for Asian people especially Indonesian people.**

Keywords. Entrance surface dose (ESD), CALdose_X, TLD, radiodiagnostic examination

Introduction

The utilization of X-rays in the health sector plays a fairly large, especially in the enforcement of illness diagnosis. Report of the United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR) stated that in last 2000 X-ray diagnostic examination was reported to reach 2,100 million examinations, or about 360 examinations for every 1,000 people worldwide. This number is 10% higher than for the period 1991-1995 where the number of examination was 330 per 1,000 population, which means there has been an increase in the number of examinations every year (UNSCEAR, 2000).

One of such applications is the use of X-rays for diagnosis of thoracic organ or known as the thorax photo. Thorax photo is one of the health requirements for both civil and private sector servants, as well as the admission of new students who requires the diagnosis of thorax photo. Thorax photo is a photo Roentgen that needed by the public.

Evaluation of these doses provides valuable baseline information for the optimisation of dose management. The data also allow Indonesian doses to be compared with those of other countries and for the setting of national Diagnostic Reference Levels (DRLs). The latter is used as indicative benchmarks for comparative radiological practices.

International Atomic Energy Agency (IAEA) has provided guidance on the level of radiation dose received by the patient on a medical examination, including diagnostic checks (IAEA, 2003). In the use of X-rays for the purpose radiodiagnostic safety and health it should be considered in accordance with the principles of radiation protection, in particular the principles of justification and optimization. In addition, the principles of radiation protection in medical exposure in particular the principle of optimization can be carried out continuously. This event will begin with a complete data collected in hospitals across Indonesia using the X-rays.

The radiation dose from X-ray is important to be considered because the effects of X-ray radiation. During this dose level, Entrance Surface Dose (ESD) of patients undergoing thoracic photo can be measured using TLD-100 dosimeters. Patient dose measurements

using TLD-100 dosimeter depends on the availability of the number of TLD-100. In order to overcome the problems in the dependence with TLD-100 dosimeters, the dose can determined by using calculations with CALDos_X 5.0 software program. This program can be used to determine the ESD of patients and other nearby organs.

Materials and Methods

This study consisted of three parts, including compliance test which was related to patient dosimetry to ensure that the X- ray machine was in good working condition (PerKa BAPETEN, 2011), output data from X-ray machine to patient dose measurements using TLD-100 dosimeters which included the estimated value of ESD, and dose determination using CALDos_x Program, Version 5.0.

Compliance Test

To assure the performance of the X-ray machine, a compliance test was performed. The parameters of the test were voltage accuracy (kV), timer accuracy (s), the linearity of the output radiation at various current conditions and time (mA.s), reproducibility of voltage, time and radiation dose, and the radiation beam quality of the X-ray, beam alignment and congruency test of X-ray machine (Anonimous, 2006). UNFORS non invasive beam analyzer was used for measurement of X-ray output.

Patient Dose Measurement

The measurement of patient dose (ESD) was done using TLD-100 LiF:Mg,Ti Harshaw which had been calibrated to X-ray source at Secondary Standard Dosimetry Laboratory in PTKMR BATAN. The TLDs were read using a Harshaw 2000AB TLD reader and annealed in Thermolyne Furnace for 1 hour at 400˚C and in an oven (Memmert Inc.) for 2 hour at 200˚C. For the measurement of ESD, the following parameters were collected for each of patient included patient name or identifier, age, sex, body height and weight, kVp, mA, exposure time, FFD, FSD, film size, X-ray machine model, type of examination and TLD's number. After that sets of 3 TLDs were placed on the patient's skin in center of the X ray beam (IAEA, 2004).

One day after irradiation the radiation response in TLD was read with TLD-Reader Harshaw 2000AB. TL intensity readings were performed twice for each TLD-100 chips used. The first reading was the reading of the total TL intensity, Mi, while the second reading was the reading of background, Mo. Net TL intensity (M) was the result of total intensity subtracted by the background intensity divided by 3 (for 3 chips of TLD-100 on each exposure) with the following formula.

$$\bar{M} = \frac{\sum_{i=1}^{3}(M_i - M_0)}{3} \quad (1)$$

Accumulated radiation dose received by TLD during the monitoring process can be calculated by multiplying the net TL intensity (after substracting by the value of TLD background reading) with a calibration factor (N_{Qk_0}) of TLD to X-rays. TLD calibration factor is defined as one per sensitivity (N_{Qk_0} = 1 / S) with units of mGy/nC. Furthermore, this dose is called the ESD. The correction factor, *kQ*, for the radiation quality and the correction factor, *kf*, that corrects for the effect of fading of the thermoluminescence signal between irradiation of the dosimeter was determined as following equation (IAEA, 2007).

$$K_e = \bar{M} N_{Qk_n} k_Q k_f \quad (2)$$

Dose Estimation Using CALDose_X

CALDose_X Version 5.0 Software Program was used to estimate the radiation dose received by the patient included entrance dose and dose received by critical organs. For calculating ESD using this program, the following parameters were put into the software which included patient name or identifier, age, sex, type of examination and its projection, kV, mAs, FFD, X-ray machine model, output value (µGy/mA.s at 1 m for some kV observed).

Results and Discussion

A. Compliance Test of X-ray

For X-ray machine used to measure the patient dose, it needed a compliance test to know the reliability / performance of X-ray machine, where some results of the compliance test were required for the input of the CALDos_X program. The compliance test results were presented in Table 1.

Table 1. Compliance Test Results of X-ray Machine.

Test parameter	X Ray A	X Ray B	Standard / Tolerance Limit [7]
kV accuracy	5.63	3.55	0.10
timer accuracy	5.30	7.49	0.10
output linearity	0.05	0.06	0.10
kV reproducibility	0.20	0.37	0.05
timer reproducibility	0.00	0.08	0.05
dose reproducibility	1.38	0.65	0.05
beam quality at 70 kV	3.02	2.37	2.10

Notes for Table 1:
*kV, timer accuracy is the value of standar deviation
*kv, timer, dose reproducibility are the value of coefficient variation
*beam quality in mmAl
*output linearity are :

$$linearity\ coefficient = \frac{X_{max} - X_{min}}{X_{max} + X_{min}}$$

ESD Measurement and Calculation Dose using CALDose_X Version 5.0 Program

Result from measurement and calculation are described in Table 2 below.

From Table 2 and 3, the ESD result using TLD and calculation by CALDose_X program showed almost similar results. But there was still a deviation between measurement and calculation. The deviation is shown at Figure 1 and 2.

The dose deviation were 0 – 0.069 mGy for X-ray A and 0.001-0.027 mGy for X-ray B. These deviation was occurred because of the uncertainty while measuring the input parameter for calculating using CALDose_X. Beside that, the CALDose_X Program had an assumption that the patient's body weight was 73 kg for male and 60 kg for female. While the patient's body height was 173 cm for male and 163 for female. In fact, the patient weight and height were different with this assumption.

Tabel 2. Patient (1st population) Dose Measurement and Calculation using X-Ray A.

No	Name	Age	Sex	kVp	Height (cm)	Weight (kg)	mAs	FFD (cm)	ESD TLD (mGy)	ESD CALDose (mGy)
1	Sl	38	Male	70	165	53	4	170	0.123	0.130
2	Mz	47	Male	77	160	70	5	168	0.211	0.211
3	Zr	53	Male	70	167	50	4	150	0.111	0.180
4	Bh	50	Male	70	155	48	4	176	0.131	0.120
5	SL	57	Male	70	153	54	4	166	0.133	0.140
6	Mh	43	Male	63	170	50	2	120	0.078	0.110
7	As	72	Male	66	165	67	2.5	121	0.180	0.160
8	HI	27	Male	73	176	75	4	136	0.264	0.250
9	Sl	24	Male	70	173	60	4	150	0.165	0.180
10	Sh	38	Male	66	175	68	2.5	163	0.087	0.087
11	Aa	67	Male	70	170	64	5	163	0.201	0.180
12	Al	64	Male	77	160	70	5	120	0.246	0.230
13	Zl	50	Male	76	120	56	3.2	155	0.133	0.160
14	Nl	26	Female	73	148	63	4	155	0.174	0.160
15	Nh	60	Female	63	148	70	12.5	164	0.355	0.310

Tabel 3. Patient (2nd population) Dose Measurement and Calculation using X-Ray B.

No	Name	Age	Sex	kVp	Weight (kg)	Height (cm)	mAs	FFD (cm)	ESD TLD (mGy)	ESD CALDose (mGy)
1	Ann	42	Male	63	16	172	68	153	0.188	0.19
2	Adh	50	Male	60	16	163	50	150	0.140	0.15
3	Olv	25	Female	60	12	156	54	150	0.103	0.11
4	Hsn	36	Male	62	16	163	70	150	0.162	0.17
5	Ags	32	Male	68	16	172	65	150	0.209	0.21
6	TnF	42	Male	68	16	178	85	150	0.216	0.21
7	Rpn	47	Female	60	16	155	60	150	0.120	0.14
8	Ngh	23	Male	65	16	172	65	150	0.163	0.19
9	Sph	66	Female	66	16	155	60	150	0.190	0.18
10	Sdg	50	Female	60	12	158	60	150	0.103	0.11

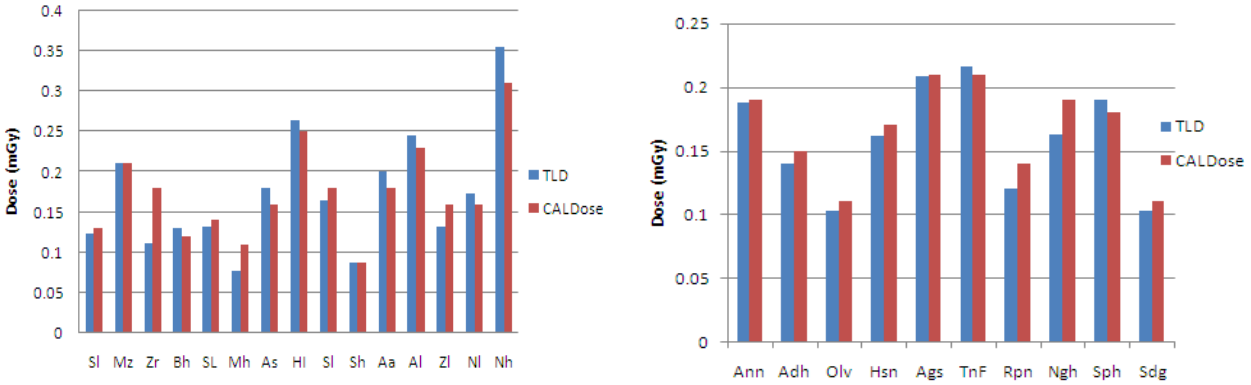

Figure. 1. Comparison of ESD using TLD and CALDose from X-ray A (left) and X-ray B (right) machines.

The correlationship between the ESD measured by TLD and ESD calculated by CALDose_X is shown in Fig 2.

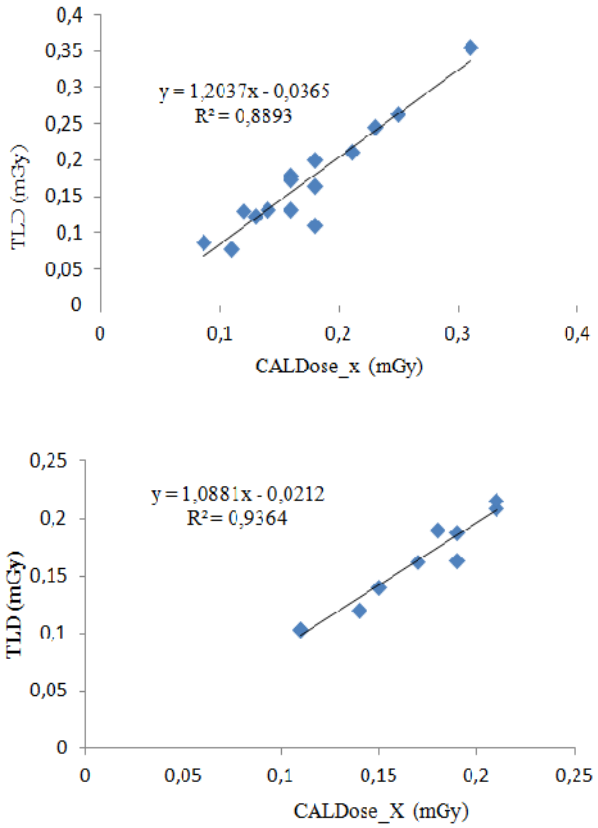

Figure 2. Correlation between ESD measurement and calculation from X-ray A (above) and X-ray B (below).

Figure 2 shows the correlation between ESD measured by TLD and ESD calculating by CALDose_X. By using a least square, the coefficient correlation were 0,8893 and 0,9364 for X-ray A and B, respectively. This means that there was a good correlation between the value of ESD measured by TLD and ESD calculating by CALDose_X. Therefore CALDose_X 5.0 software could be used to estimate the entrance dose received by patient who undergoing radiodiagnostic examination since there was a good correlation between measurement using a TLD and calculating using a software. But the calculated ESD was not same with the ESD from TLD. It should be substracted by a value according to the equation above.

Conclusions

From the results it can be concluded that the estimation dose by using CALDose_X 5.0 Software was reliable to complement the use of TLD. However, it need a further research to obtain the dose estimation method which appropriate for Asian people especially Indonesian people.

Acknowledgements

Thanks to those who have helped this research included Zainul Abidin from Banda Aceh Hospital and Pertamina Hospital in Balikpapan.

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The Accepted Radiation Dose of Pediatric Patients in Diagnostic Radiology Examinations using LiF:Mg,Cu,P

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Abstract. **The radiation dose levels to be encountered should be as low as reasonably achievable. According to UNSCEAR and ICRP 60 infants and children estimated contribute 10% of the total number of radiological examinations, and having a greater risk than adults. Therefore, to measure accurately low-dose radiation to pediatric patients, it is necessary to have a dosimeter which has a high sensitivity and its fading can be neglected. In this experiment TLD LiF:Mg,Cu,P from Harshaw product was used. Since the reference doses were not yet available and considering the level of radiation risk, the measurement of the radiation dose received by Indonesian children in diagnostic radiology examination is necessary to be done. The measurement of radiation dose of pediatric patients in radiology facilities in some hospitals have been done with grouped according to age 0, 1, 5, 10 and 15 years old. Dose measurements were carried out for the examination such as AP/PA/LAT Thorax, AP/PA Skull, AP/LAT abdomen, and measurement of extremity dose. The measurements result of effective doses received by pediatric patients 1 year old for abdomen type examination were generally higher compared to data issued by NRPB, and relative lower for another. For other examination types, the minimum values of radiation dose obtained were lower than NRPB data. In the examination of AP/PA thorax, the dose data provided by NRPB was 80μGy, 110μGy and 70μGy for the age of 1, 5 and 10 years old, respectively. The results of dose measurement for the same examination type were 78 μGy, 117.1 μGy and 155 μGy, respectively, with min and max values were (8.9 – 174.0) μGy, (37.2 – 369.2) μGy and (8.4 – 267.4) μGy. The various results may be caused by a lack of medical physicists in diagnostic radiology facilities, lack of skills using dosimeters and/or human error. To solve this problem, the compliance test must be done regularly and continuously. In addition, there should be training for the X-ray machine operator.**

Keywords: radiation dose, pediatric patients, high sensitivity dosimeters, LiF:Mg,Cu,P

Introduction

Many studies have been conducted to determine the value of received dose in medical X-ray examinations and its correlation with risk level probability of exposure from ionizing radiation which depends on the different ages (or size), especially for pediatric patient has increased in recent years (Armpilia et al., 2002; Hintenlang et al., 2002; Suliman and Elshiekh, 2008; Brindhaban and Eze, 2006; Mohamadain and Azevedo, 2009; Ogundare and Ajibola, 2004; Schneider et al., 2000). The International Commission on Radiological Protection (ICRP) has been recommended to use diagnostic reference levels (DRLs) as first step for optimization of radiation dose in diagnostics radiology (ICRP, 1996). International Atomic Energy Agency (IAEA) also recommended that the guidance levels or DRLs for medical exposures shall be established for use by medical practitioners and relevant professional bodies in consultation with the regulatory authority to ensure optimized protection of patients and maintain appropriate level of good practice (IAEA, 2003).

Pediatric patients are more sensitive to ionizing radiation compared with adult patient, estimated 10% of the total number of radiological examinations, and having 2 to 3 times higher level risk of developing a radiation-induced cancer, hereditary effects or other serious disorders, because of their greater cell proliferation rate and long life span expectancy (UNSCEAR, 2000; ICRP, 1990), whereas Rassow et al. (2000) stated that the radiation risk (both stochastic and deterministic) is 4-8 times higher for pediatric patients than for adults. This range of sensitivities encompasses both deterministic and stochastic effects. Low doses radiation on pediatric patients can be increase of risk, nevertheless, radiological examinations should be performed with consideration of the potential risks of radiation exposure must be balanced against the diagnostic information and also should be kept as low as reasonably achievable (ALARA principle) (IAEA, 1990). The measurement of low-dose radiation with accurate and precision on pediatric patients in diagnostic radiology examination is required which has a high sensitivity dosimeter and its fading can be neglected.

According to Bower and Hintenlang (1998), the ideal dosimeter for monitoring surface radiation during procedures should: (1) has an extremely small area to prevent interference with the image quality; (2) has a response which is linear within the measured dose range; (3) be simple to use; (4) provide real-time outputs; (5) demonstrate small angular response; and (6) be tissue equivalent. Dosimeters that are widely used in medical dosimetry are thermoluminescence dosimeters

(TLD). For many years, TLD of lithium fluoride (LiF:Mg,Ti or TLD-100) chips doped with magnesium and titanium (Harshaw) has been dominate as TLD in medical dosimetry for measuring dose radiation on diagnostics radiology examination. In recent years, however, a relatively new material and 23 more sensitive (Sofyan and Kusumawati, 2012), TLD LiF with activator Mg, Cu and P (LiF:Mg,Cu,P or TLD-100H), has attracted attention and started being used in medical dosimetry replacing the TLD-100, especially at low radiation dose monitoring (Glennie, 2003).

Although there are recommendations of ICRP and IAEA to apply the DRLs, however it cannot be used directly for all countries because of differences in patient physically. In Indonesia, the reference doses are not yet available, and with considering the level of radiation risk for radiation safety, then the measurement of the radiation dose received by Indonesian children in diagnostic radiology examination is necessary to be done. Further, one of the main benefits from patient dose surveys is to provide information on doses in different hospitals for comparison, which give an indication whether the techniques used in our own hospitals are optimized in terms of dose. Furthermore, it is expected to main benefits of activities in different hospitals was to determine and compare techniques that be used in diagnostic radiology examination, as well as optimizing the dose received pediatric patients.

Materials and Methods

TLD-100H (LiF:Mg,Cu,P) chips Harshaw/ Thermo Scientific product with dimensions of $3.2\times3.2\times0.89$ mm^3, effective atomic number (Z_{eff}) 8.14 and linearity of dose response 10^{-6} to 10 Gy, as well as the annealing temperature that is recommended at 240 °C for 10 minute (Zoetelief, 2000) were used in this research. The response of TLD is read by using Harshaw TLD Reader Model 2000A/B which flowed nitrogen gas at pressure of 20 psi to minimize the TL signal derived from non-ionizing radiation. To get optimal reading, the voltage of photomultiplier tube device was set at 617 volts, so that sensitivity become 169 nC/sec, integrated TL signal for 25-30 seconds and the maximum temperature in the range of 240-250°C. The heating rate will affecting the intensity of the TL, therefore it was set at 7°C/sec (Bos, 2007). Room temperature (< 24.5°C) and humidity (~ 31%) during the reading of the TLD is also concerned so that the room can be conditioned properly and optimally. Due to thermal stimulation on TLD-100H can cause the thermal quenching effect and decrease of sensitivity, the annealing process is performed for 10 minutes at temperature of 220°C below the recommended temperature, then was proceed with packaging of each packet consist of 3 TLD-100H.

The research of radiation dose acceptance was conducted in several hospitals in Jakarta, Bandung and Yogyakarta which has facilities of diagnostic radiology examination for pediatric patients. Selected hospitals based on number of pediatric patients who perform diagnostic examinations with interval of ages between 0-15 years in five groups. To facilitate analysis and discussion, the grouping of pediatric patients based on patient age is equated to patient data from National Radiological Protection Board (NRPB), namely pediatric patients 0 years (up to 1 day), 1 year (> 1 day-1 year); 5 years (> 1-5 years); 10 years (> 5-10 years) and 15 years (> 10-15 years).

The measurement of radiation dose in pediatric patients for each examination was carried out with placing the TLDs in radiation field to be measured. One packet of TLDs is for one-time exposure. The exposed TLDs to X-ray radiation will be read with using TLD Reader Harshaw model 2000A/B. The value of radiation dose received by pediatric patients was average readings reduced with background and then multiplied by a correction factor. Although in this study, the patient data recorded include the child's name, age, weight, sex, type of examination, kVp, mAs, but the values of the dose received by pediatric patients is become priority to known. This research activity is expected to be the first step to make DRLs especially for pediatric patients on diagnostic radiology examination in Indonesia.

Results and Discussion

The use of ionizing radiation in radiology field for various medical purposes needs to consider aspects of risk and benefit to be achieved. The evidence suggests that radiation dose exposure of peoples received providing contribution in total radiation significantly, whether from natural radiation or artificial sources of radiation. Therefore it is necessary to set guideline that limits radiation dose for each type of exposure in radiodiagnostic. Different from radiation doses to workers and public, the medical radiation dose limit value can not be determined, because there are other factors that must be in accordance with expected diagnostic purposes. In Indonesian Government Regulation it also expressed that the radiation dose limitation for patients of radiodiagnostic examination do not applicable (PP-RI no. 37, 2007). Thus, exposure restrictions to protect pediatric patients can only be given in the form of limit values as guide for implementing of examination type using radiodiagnostic techniques.

The studies of patient radiation doses arising from diagnostic radiology examinations have largely concentrated on the adult population. In pediatric patients, the generally major problem in data collection for is the various ranges of pediatric patient sizes within any age. In addition, difficulties arise in the examination of pediatric patients are also be a factor that can not be ignored and needs serious attention. However, research to determine the radiation dose received for pediatric patient must be done, especially to collect and make the data of DRLs for Indonesian pediatric patient.

As the result of study, the effective dose received by pediatric patients for each examination type such as thorax, abdomen, cranium, vertebrae and extremities examination is given in Figure 1. While the Table 1 are values of potential energy (kVp) based on age of pediatric patient and type of examination. In this table can also be viewed comparison of data acquisition with pediatric patient dose data from NRPB. The effective doses received pediatric patients 10 year old for abdomen type examination are generally higher compared to data issued by the NRPB, and relative lower for another age of patient. The minimum values of radiation dose obtained from research for each X-rays exposure in all examination types are lower than NRPB data. Meanwhile, on examination of the AP/PA thorax, the dose data provided by NRPB were 80μGy, 110μGy and 70μGy respectively for the age of 1, 5 and 10 years. The results of dose measurement for same examination type were 78 μGy, 117.1 μGy and 155 μGy, respectively. With values of min and max are (8.9 – 174.0) μGy, (37.2 – 369.2) μGy and (8.4 – 267.4) μGy. Until now NRPB dose not have any data cranium examination for children aged 10 and 15 years, so that the data obtained from this reserach can not be compared with data NRPB.

The radiation dose received to pediatric patient depends on kVp and mAs parameters given by operator of X-ray machine. The variation of kVp and mAs values was influenced on differences of size and age of pediatric patients. In general, because of every operator prefer the good image is formed on film, often does not consider the value of radiation dose that should be given to pediatric patient. Thereby going considerably, the operators will be more attention to the principle of justification and not to consider the principle of optimization. This is related to the better quality of the image formed on the film, because it will make it easier for doctors to analyze a disease.

In the last few years of radiographic film is gradually being replaced with Computed Radiography (CR) cassette. CR cassette which utilizes a computerized system, providing much convenience in editing image is formed that is not too dark or too light, so the film has been printed with good image quality according medically. Indirectly, it can reduce the number of films rejected due to the image that are too dark, too light, no focus and double shadows. Although many advantages of CR cassette, as more tolerant of under/over exposure, but there is still a lack of uniformity of the dose received per pediatric patient in same age as shown in Figure 1. This possibility can be caused by operators who are still not using dose reference levels (DRL). In radiation protection, radiation dose received on pediatric patients needs to get more serious attention. Because the measuring radiation dose in radiology for pediatric patients often encountered difficulties. Some difficulties for the measurement of dose are patient movement, dosimetric techniques should be specifically, non-cooperative pediatric patient, image for the assessment of positioning (symmetry and absence of tilting etc), incorrect positioning can cause of inadequate image quality and inferior image quality. The patient movement is a big problem in pediatric patient and would require specific adjustment in radiographic techniques.

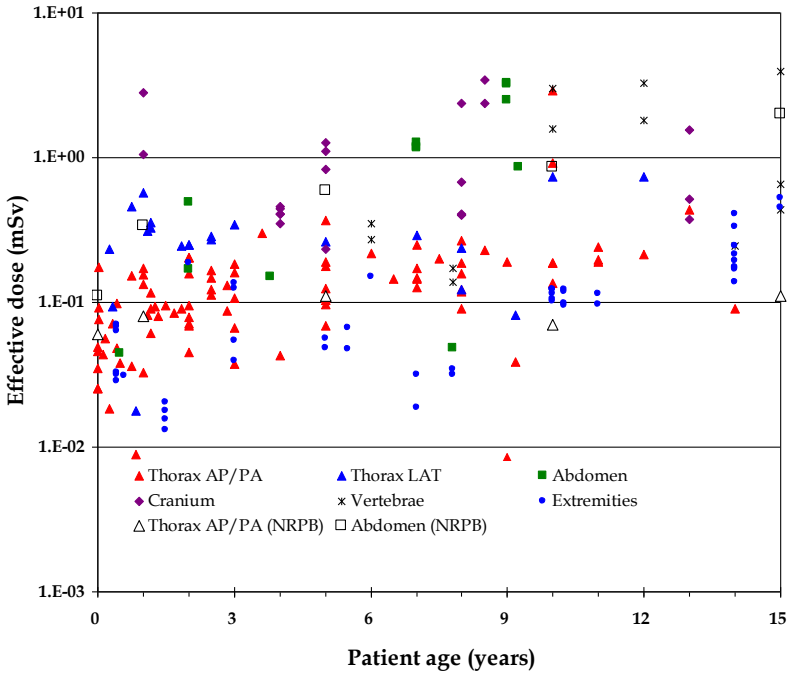

Figure 1. Radiation dose received in pediatric patient (age 0-15 years) for thorax, abdomen, skull, vertebrae and extremities examinations.

Table 1. Accepted effective doses in pediatric patient for thorax, abdomen, and skull examinations

No	Examination Type	Age	Potential Energy (kVp			Dose (μGy)			NRPB*)
			Min	Max	Mean	Min	Max	Mean	
1.	Abdomen	0	-	-	-	-	-	-	110
		1	-	-	-	-	-	-	340
		5	51.00	60.00	54.000	149.3	492.7	270.3	590
		10	48.00	77.00	61.125	47.9	3,253.9	1,688.6	860
		15	60.00	66.00	63.000	408.8	445.8	427.3	2,100
2.	Thorax AP/PA	0	43.00	45.00	44.333	25.3	49.2	36.4	60
		1	40.00	60.00	47.842	8.9	174.0	78.0	80
		5	34.00	99.00	49.683	37.2	369.2	117.1	110
		10	39.00	63.00	50.773	8.4	267.4	155.0	70
		15	42.00	65.00	53.714	90.9	437.8	209.4	110
3.	Cranium AP/LAT	0	-	-	-	-	-	-	-
		1	-	-	-	-	-	-	600/340
		5	44.00	73.00	61.818	235.1	2,816.9	827.6	1,250/580
		10	48.00	79.00	64.000	403.8	3,420.6	1,610.1	-
		15	62.00	70.00	65.333	376.0	1,563.3	819.2	-

*) NATIONAL RADIOLOGICAL PROTECTION BOARD, Doses to Patient from Medical X-Ray Examinations in the UK: 2000 review, NRPB-W14, Chilton (2002).

Conclusions

The results of measurements for thorax examinations, the min to max values of dose accepted for 1, 5 and 10 years old of pediatric patients were (8.9 – 174.0) μGy, (37.2 – 369.2) μGy and (8.4 – 267.4) μGy, respectively. Meanwhile, on thorax examination, the dose data provided by NRPB were 80μGy for 1 year, 110μGy for 5 years and 70μGy for 10 years. In general, the minimum value of measurement results for all types of examination (thorax, abdomen and Cranium) is lower than NRPB data on the same examination. But the maximum dose received by pediatric patients is very high compared to the value of NRPB. The various results may be caused by a lack of medical physicists in diagnostic radiology facilities, lack of skills using dosimeters and/or human error. To solve this problem, the compliant test must be done as regularly and continuously. In addition, there should be training for the X-ray machine operator.

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DISCUSSION

Question:
Iin Kurnia, PTKMR-BATAN:
It is only a suggestion that the picture of person/children you presented in the power point should be covered his/her eyes.

Answer:
Thank you for your suggestion. The next presentation I will close the identity of patient with closing the eyes.

The IAEA/SSDL Comparison of Therapy Level of Ionization Chamber Calibration Coefficients in Terms of N_k (Air Kerma) and $N_{D,w}$ (Absorbed Dose to Water) For ^{60}Co Beam

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Abstract. **At present, routine calibrations of radiotherapy dosimeters at the SSDL Jakarta are performed in ^{60}Co beams in terms of air kerma (N_K) and absorbed dose to water ($N_{D,w}$). In 2012 the SSDL Jakarta participated in the IAEA/SSDL comparison of therapy level ionization chamber calibration coefficients in terms of N_K and $N_{D,w}$ for ^{60}Co beam. The aim of this comparison was to get the verification about the SSDL's ability to carry out calibrations within acceptable limits and proper traceability exists to primary standards. The SSDL should check the stability and calibrate a transfer chamber in terms of N_K and $N_{D,w}$ using the IAEA or its own procedures before and after the chamber calibrated by the IAEA. Then it is sent to the IAEA to be calibrated. A transfer and A reference chambers used were 0.6 cc chamber NE 2571 s/n 2491 and s/n 2693. Both chambers were connected to an electrometer NE 2570/IB s/n 1319. Then the transfer chamber is calibrated in terms of N_K and $N_{D,w}$ by using ^{60}Co beam. Calibration in term of N_K was performed at SCD of 80 cm and FS of 10 cm x 10 cm. Meanwhile, calibration in term of $N_{D,w}$ was carried out in a water phantom at source to SSD of 80 cm and FS of 10 cm x 10 cm and 5 mg.cm^{-2} depth. The means of N_K and $N_{D,w}$ calibrated by the SSDL were (41.16 ± 0.56) mGy/nC and (45.23 ± 0.48) mGy/nC . Meanwhile, N_K and $N_{D,w}$ given by the IAEA were (41.27 ± 0.56) mGy/nC and (45.12 ± 0.48) mGy/nC. The ratios for N_K and $N_{D,w}$ between the SSDL and the IAEA were 1.000 and 1.000, respectively. The results were very satisfied because the deviation equal to or less than 1.5% and was classified as minor.**

Keywords: comparison, ionization chamber, air kerma, absorbed dose to water

Introduction

Prior to 1999, reference dosimetry of clinical high-energy photon beams was largely performed using chambers calibrated free in-air in terms of air kerma, N_K, at reference radiation quality. There has been a world-wide development of calibration techniques directly in terms of absorbed dose to water, $N_{D,w}$, and replacing N_K by $N_{D,w}$ as the calibration quantity.

Calibration in absorbed dose to water will be performed in a water directly at different high-energy beam qualities against absorbed dose to water standards. However , due to the good stability and the world-wide easy availability of ^{60}Co gamma beams for calibration, more practical approach is being used which relies on direct absorbed dose to water at ^{60}Co beams, with subsequent evaluation of calibration coefficients for other beam qualities using an appropriate radiation quality correction factor.

In the last few years in Primary Standard Dosimetry Laboratory (PSDL) therapy level ionization chamber calibration coefficient had been changed from N_K to $N_{D,w}$. This change was intended to determine base of dosimetry in connection with absorbed dose to water in clinical field use (Andreo, 1992; Rogers, 1992). At present, routine calibrations of radiotherapy dosimeters at the SSDL Jakarta are carried out in ^{60}Co gamma beams in term of N_K and $N_{D,w}$.

In 2010 the SSDL Jakarta also participated in the APMP Comparison of radiotherapy dosimeters calibration coefficient in term of N_K and $N_{D,w}$ but the chamber calibrated was belonged to PSDL Taiwan (Lee, 2010). In 2012 the SSDL Jakarta also participated in the IAEA/SSDL Comparison of therapy level ionization chamber calibration coefficients. This comparison was followed by all SSDLs which had become a member of the IAEA/SSDL network. In this comparison, the chamber being calibrated was a chamber used as field class ionization chamber in the SSDL Jakarta. In this comparison the SSDL Jakarta used dosimeter having a chamber coefficient calibration traceable to the ARPANSA PSDL.

The aim of the SSDL Jakarta following this comparison is to get the verification about the SSDL's ability to carry out calibrations within acceptable limits and proper traceability exists to primary standards. It will be of high importance to collect experiences on this calibration technique in order to evaluate the impact of the new method on clinical dose determination, i.e. on the dose delivered to the patient.

This paper describes comparison and calibration methods, the results of comparison of the chamber belonged to the SSDL Jakarta.

Comparison Method

General requirements should be fulfilled by an SSDL that is willing to participate in the comparison for therapy level are that the SSDL has :

1. Therapy level ^{60}Co gamma beam irradiation device or access to a ^{60}Co therapy unit at a hospital;
2. Therapy level working standard ore reference standard ionization chamber and electrometer routinely used in calibrations;
3. Devices to set precisely the field size and the source to chamber distance and to position the phantom in the case of absorbed dose to water;
4. Barometer, thermometer and hygrometer;
5. Field class ionization chamber with build-up cap that belongs to the chamber design;
6. Check source for use with the above mentioned field class ionization chamber; and
7. A water phantom with minimum dimensions of 30 cm x 30 cm x 30 cm and a PMMA water sleeve for the chamber.

It is emphasized that only a field class ionization chamber should be used as a transfer chamber in this comparison, and not the chamber used as the reference or working standard of the SSDL. In this case the field class ionization chamber to be used in the comparison should be one of the types as mentioned in the IAEA Technical Reports Series 277 (IAEA, 1987). A working standard is used for the measurement of air kerma or absorbed dose to water, while the reference standard could be used if the SSDL has only one standard class chamber.

The two comparison methods given by the IAEA to be performed by the SSDL are comparison of air kerma calibration coefficient and comparison of absorbed dose to water calibration coefficient. For reference condition that is used in two comparison methods can be seen in Table 1.

Procedures used in this comparison

1. Procedures at the SSDL prior to sending the chamber to the IAEA that should be performed.

   First, the transfer and reference standard chambers are checked their long term stability. After they are stable, the transfer chamber is calibrated in terms of N_K and $N_{D,w}$ in the SSDL's ^{60}Co beam in accordance with the normal calibration procedure applied at the SSDL. The readings or charges of the transfer chamber calibrated and the reference standard chamber are corrected for the ambient conditions (20^{o}C and 101.3 kPa). The relative humidity in the room at the time of calibration should be checked. Then the calibration coefficient, the estimated uncertainty and information on the SSDL's standards on the reference standard in case the SSDL has only one standard chamber are recorded in the worksheet. Finally, the SSDL send the calibrated chamber with a copy of the worksheet (the original of the worksheet is kept at the SSDL for reporting the results of the second calibration to the SSDL officer shortly after calibration).

   In case the SSDL uses a calibration coefficient that is derived by calculation from air kerma calibration coefficient for getting $N_{D,w}$, the SSDL must indicate in the worksheet which code of practice of dosimetry is used for the calculation.

2. Procedures at the SSDL upon arrival of the returned chamber from the IAEA First as soon as possible after the return of the transfer chamber to the SSDL the chamber is checked for transport damage and performed the long term stability of the calibrated chamber. Then the SSDL compares the results with that of the stability check made prior to sending the chamber to the IAEA. If the discrepancy in the reading exceeds the SSDL's internal action level the SSDL must notify the IAEA. After that, the SSDL must calibrate the chamber again as above and record the results. Then the SSDL compares the results with that of the calibration made prior to sending the chamber to the IAEA. Finally, the SSDL must report all results using the original worksheet.

Reporting the results by the IAEA

The IAEA will transmit the results to the SSDL Jakarta only after the results of the second calibration have arrived at the IAEA. In case the results action level for major deviation, the IAEA will contact the SSDL immediately. The SSDL should send a copy of worksheet contained the calibration results in terms of N_K and $N_{D,w}$, estimated

Table 1. The Reference condition used in two comparison methods.

	Comparison of air kerma calibration coefficient	**Comparison of absorbed dose to water calibration coefficient**
Reference condition :		
Source to chamber distance	100 cm	100 cm
Field size	10 cm x 10 cm	10 cm x 10 cm
Chamber reference point for calibration	Chamber geometrical center	Chamber geometrical center
Relative humidity	30% - 70%	20% - 80%

uncertainties, and the long term stability check and other information on the reference standard chamber together with the transfer chamber to the IAEA. After the IAEA returns the chamber to the SSDL, the SSDL send the original worksheet that has been completed with the calibration results of the transfer chamber

Action levels

In order to ensure conformity among the SSDL members, it is necessary to verify that the SSDLs perform calibrations within reasonable tolerances. The percentage deviation from unity of the ratio of the calibration coefficient determined the percentage deviation the SSDL and the IAEA shall be used to establish an action level. This percentage deviation must be equal to or less than 1.5%. Deviations larger than 1.5% will be classified as major. One of the duties of being full member of SSDL network is to perform this comparison successfully. Deviation equal to or less than 1.5 % will be classified as minor and will be report to the SSDL for their own analysis and follow up. The value of 1.5 % was established based on data taken from, TRS 374, table IV (IAEA, 1994). Consequently, using a coverage factor of k = 2, the action level was set at the value 1.5%.

The IAEA standard traceable to those of the BIPM. SSDLs whose standards are traceable to other PSDLs may show deviation as a result of the difference between their primary standards. The published values of the difference between standards of the BIPM and the particular PSDL used by the SSDL will be taken account when computing the ratio between the SSDL and the IAEA.

Uncertainty budget calculation

The SSDL must calculate the uncertainty of calibration coefficient both N_K and $N_{D,w}$ performed as type A and Type B as published in the International Organization for Standardization (ISO) in 1995 on " Guide to The Expression of Uncertainty in Measurement" (International Organization for Standardization, 1995). It is expected that the SSDL combined standard uncertainty will be about 0.8 % at the level of one standard deviation. The additional uncertainty due to calibrating the SSDL's chamber at the IAEA is not expected to increase the uncertainty in the ratio significantly. The SSDL can determine what factors must be included in the uncertainty calculation. The final calculation results of uncertainty when calibration before and after the calibration performed by the IAEA are included in the worksheet.

Materials and Methods

Long term Stability of the Ionization Chamber

Before the transfer and reference standard chambers were calibrated, it is necessary to to know the stability of both chambers by performing a long term stability check of both chambers. A chamber NE 2571 serial number 2693 as a reference standard chamber was connected to a Farmer electrometer NE 2570/1B. Then the chamber was placed inside the ^{90}Sr source so was thermometer. To achieve an electronic equilibrium the chamber was irradiated with ^{90}Sr for 300 seconds. Then the chamber was irradiated for 250 seconds and the charge / reading and reference condition were recorded. After that the chamber was irradiated again for 250 seconds respectively until 5 data were obtained. The deviation in percentage is defined as the ratio between the reference reading corrected by a decay factor , M_{ref} , minus the reading at the time of checking, M , and the reference reading corrected by a decay factor (IAEA, 1970).

$$\text{Deviation} = \frac{M_{ref} - M}{M_{ref}} \times 100\ \% \quad \text{.....(1)}$$

The deviation should not exceeds of ± 1%. If the deviation was more than ± 1%, the chamber could not be used for measurement.

The transfer chamber NE 2571 s/n 2491 was connected to the same electrometer as the reference standard chamber used. By using the same steps above the chamber was irradiated using the same ^{90}Sr source as the irradiation of the reference standard chamber. The set up of stability check and both chamber used in this comparison could be seen on Figure 1and Figure 2.

Figure 1. The set up of long term stability check of the chamber.

Figure 2. Two chambers used in this comparison. The transfer chamber on the right side and the reference standard chamber on the left side.

Calibration in term of Air Kerma

The reference standard chamber that had been connected to the electrometer was placed in air at the source to chamber distance (SCD) 80 cm and field size (FS) of 10 cm x 10 cm. To achieve an electronic equilibrium the chamber was irradiated with ^{60}Co gamma beam from Alcyion/ Cirus ^{60}Co therapy unit having ^{60}Co source type of COT-20 with activity of 233,636 TBq on 1 June 1999 for five minutes. Then the chamber was irradiated for 1 minute and the charge / reading and reference condition were recorded. After that the chamber was irradiated again for one minute respectively until 5 data were obtained.

By using the same steps above the transfer chamber that had been connected to the same electrometer as the reference standard chamber used was irradiated using the same ^{60}Co beam as the irradiation of the reference standard chamber. The set up for calibration for the air kerma calibration coefficient could be seen on Figure 3.

Figure 3. The setup for calibration of transfer chamber and the reference condition for the air kerma calibration coefficient (SCD 80 cm and FS 10 cm x 10 cm).

Calibration coefficient in term of N_K is defined as the product of reference reading corrected to the P and T, M_{ref} and the calibration coefficient of reference chamber, $N_{K\ reference}$, divided by the calibrated chamber reading corrected to the P and T, M :

$$N_K : \frac{Mref .xN_{K\ ref}}{M} \quad \text{..... .....(2)}$$

Calibration in term of Absorbed Dose to Water

The reference standard chamber that had been connected to the electrometer was placed in a water phantom at the source to surface distance (SSD) 80 cm and field size (FS) of 10 cm x 10 cm. Then the chamber was placed in 5 cm depth in water using a sleeve. To achieve an electronic equilibrium the chamber was irradiated with ^{60}Co gamma beam from the same ^{60}Co beam for five minutes. Then the chamber was irradiated for 1 minute and the charge / reading and reference condition were recorded. After that the chamber was irradiated again for one minute respectively until 5 data were obtained.

By using the same steps above the transfer chamber that had been connected to the same electrometer as the reference standard chamber used was irradiated using the same ^{60}Co beam as the irradiation of the reference standard chamber. The set up for calibration for the absorbed dose to water calibration coefficient could be seen on Figure 4.

Figure 4. The setup for calibration of transfer chamber and the reference condition for the absorbed dose to water calibration coefficient (SSD 80 cm, FS 10 cm x 10 cm and 5 cm depth in water).

Calibration coefficient in term of $N_{D,w}$ is defined as the product of reference reading corrected to the P and T, $M_{reference}$, and the calibration coefficient of reference chamber, $N_{D,w\ reference,}$ divided by the calibrated chamber reading corrected to the P and T, M :

$$N_{D,w} = \frac{M_{reference}.N_{D,W\ reference}}{M} \quad \text{..... (3)}$$

Results and Discussions

Calibration Coefficient in terms of N_K and $N_{D,w}$

The calibration results of the transfer chamber in terms of N_K and $N_{D,w}$ that are calculated based on calibration factors of reference standard chamber traceable to the ARPANSA PSDL can be seen in Table 2.

From the Table 2 it can be seen that calibration factors in terms of N_K and $N_{D,w}$ that are obtained before and after calibration at the IAEA have a slight difference of 0,1 % .and 0,3% respectively. To cover this difference, the IAEA takes the mean of calibration coefficients of N_K and $N_{D,w}$.

$N_{D,W}/N_K$ Ratio

According to some references it is stated that $N_{D,w}/N_K$ ratio for ^{60}Co beam for a number of chambers used in TRS No. 398 could be seen in Figure 5 (IAEA, 2000).

From the Table 2 ratio obtained by taking the means of N_K and $N_{D,w}$ before and after calibration at the IAEA is 1.099 and according to the Figure 5 for the chamber NE 2571 the $N_{D,W}$ / N_K ratio is less than 1.10 This means that the calibration results in terms of N_K and $N_{D,w}$ were good.

Table 2. Calibration results of the transfer chamber in terms of N_K and $N_{D,w}$

The transfer chamber calibration	N_K (mGy/nC)	$N_{D,w}$ (mGy/nC)	$N_{D,w}/N_K$ ratio
Before the chamber sent to the IAEA	41.14 ± 0.56	45.30 ± 0.48	1,100
After the chamber return to the SSDL	41.18 ± 0.56	45.16 ± 0.48	1.097

Uncertainty budget calculation

Components that contribute to the measurement uncertainty of calibration coefficient include source position, the chamber calibration certificate from ARPANSA, stability of the instrument, mean reading, resolution of the instrument, thermometer and barometer calibration certificates, thermometer and barometer resolution, chamber position in air and in the water phantom. From Table 2 it can be seen that the expanded uncertainties of the N_K and $N_{D,w}$ calibration coefficients are 1.37% and 1.07%, respectively for 95% confident level.

Results of the Comparison

The result of the comparison giving by the IAEA can be seen in Table 3.

The SSDL Jakarta establishes traceability to the ARPANSA PSDL and not based its measurement standard on those of the IAEA (or of the BIPM to which the IAEA is traceable). Consequently, it is necessary to account for any difference between the particular standard at the PSDL used by the SSDL Jakarta and the corresponding standard of the IAEA. The correction factor $k_{PSDL/BIPM}$ mentioned in Table 3 is the ratio of the calibration coefficient determined at the PSDL and at the BIPM. Therefore, the result of the comparison between the SSDL Jakarta and the IAEA will be divided by the $k_{PSDL/BIPM.}$ The values of used for the correction are those published at the BIPM key comparison database (BIPM, 2013).

The ratios of calibration coefficients in terms of N_K and $N_{D,w}$ stated by the SSDL Jakarta and the IAEA are 1.00. This means that the calibration coefficients in terms of N_K and $N_{D,w}$ performed by the SSDL are the same as those performed by the IAEA. This ratio is meant to be used as a criterion to judge the metrological quality of the calibration performed by the SSDL Jakarta.

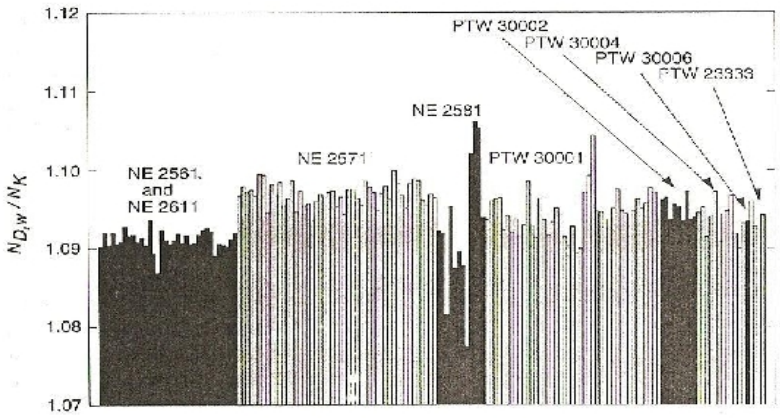

Figure 5. $N_{D,w}$/ N_K Ratio for ^{60}Co beam for a number of Ics.

Table 3. The result of comparison giving by the IAEA.

The SSDL Jakarta stated Calibration coefficients* (mGy/nC)	The SSDL Jakarta stated traceability to PSDL	The IAEA stated calibration coefficients** (mGy/nC)	$k_{PSDL/BIPM}$	the SSDL Jakarta*** ---------- IAEA (corrected)
$N_K = 41.16 \pm 0.56$	ARPANSA 2007	$N_K = 41.27 \pm 0.33$	0.9974	1.000
$N_{D,w} = 45.23 \pm 0.48$	ARPANSA 2007	$N_{D,w} = 45.12 \pm 0.45$	1.0024	1.000

*) The SSDL Jakarta stated calibration coefficient is the mean of the two coefficients, one determined prior to sending the chamber to the IAEA and the other after its return to the SSDL. The uncertainties on the calibration coefficients (k=2) as reported by the SSDL Jakarta.

**) The relative uncertainty (k=2) of the IAEA calibration coefficients is 0.8 % and 1.0 % for N_K and $N_{D,w}$, respectively.

***) Corrected ratio between 0.985 and 1.015 are considered acceptable.

Conclusion

From the result and discussion above it can be concluded that the metrological quality of the calibration performed by the SSDL Jakarta is satisfied and the SSDL Jakarta PTKMR BATAN can carry out calibration within acceptable limits and proper traceability exists to primary standards. It is suggested that the SSDL Jakarta always follow this comparison to increase and keep the metrological quality of the calibration performed by the SSDL

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The BIPM key comparison database (2013), http://kcdb.bipm.org/(BIPM.RI(1)-K1 and BIPM. RI(1)-K4.

DISCUSSION

Dadong Iskandar, PTKMR BATAN :

Question:
Why IAEA inform the laboratory if there is any major deviation?

Answer:
Because the IAEA gives the tolerance of <= 1.5% different. This calibration coefficient will affect the absorbed dose to water value measured by medical physicist. If the calibration factor was incorrect it would danger for the patient. Therefore if the deviation between IAEA and SSDL was >= 1.5%, the SSDL will be told and suggested to investigate the data, including the readings of both transfer and reference chambers, correction for air density (T,P,RH) condition calibration (SSD, FS, depth in water) and others that are included in the uncertainty calculation.

Assessment of Modern Radiotherapy Techniques for Breast Cancer Treatment in Indonesia

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Abstract. **Breast cancer has been delineated as women's cancer due to the increasing of the incidence of this cancer in women around the world. In Indonesia, breast cancer has placed as the first rank with 36.2/100,000 incidents. Another data from GLOBOCAN 2008 reported that the estimated incidence for breast cancer in Indonesia is 25.5% of total cancer incidence, with mortality rate 19.2 % of total cancer mortality. Since breast cancers can be cured in early stage, an establishment of cancer management should be done by promoting radiotherapy as one of treatment options. It is important to assess the modern radiation therapy techniques which can improve treatment outcomes in breast cancer management, especially in developing countries, such as Indonesia. The main objective of this article, is to assess modern radiotherapy techniques which are clinically used for breast cancer treatment, and to assess the possibility of the technique to be applied as an option for treating the breast cancer in Indonesia. The assessment method has been done by literature study and found that there are few options in modern radiotherapy such as brachytherapy, external beam radiation therapy and charged particle therapy. However, Intensity Modulated Radiation Therapy (IMRT) and Volumetric Modulated Radiation Therapy (VMAT), as a form of external beam radiation therapy can be used as the best option of favorable techniques for breast radiotherapy in Indonesia due to it's abillity to reduce the radiation dose delivered to the organs at risk. In addition, brachytherapy might become an option in the future.**

Keywords: breast cancer, radiotherapy, modern radiotherapy modalities

Introduction

The world's statistic of cancer in women has shown that breast cancer is the most common incidence in women. As a consequence, breast cancer has been delineated as women's cancer (Ludwig H, 1994). In Indonesia, breast cancer has placed as the first rank with 36.2/ 100000 incidents (Georgakilas, 2012). Another data from GLOBOCAN 2008 reported that the estimated incidence for breast cancer in Indonesia is 25.5% of total cancer incidence, with mortality rate 19.2 % of total cancer mortality (Ferlay et al, 2010). For this reason, the awareness about the fighting to breast cancer should be established by promoting early stage diagnosis, and followed by the right option in cancer management (Kimman M, 2012).

A cancer management is commenced when a patient has been diagnosed having cancer and continued by treatments, such as palliative surgery, radiation therapy and chemotherapy (WHO, 2012). However, from the available treatment methods, radiation therapy (radiotherapy) has been acknowledged as a significant method to kill cancer cell solely, and /or together with hormone therapy or chemotherapy or surgery (Barton et al., 2006). Radiation therapy or radiotherapy has been defined as a medical therapy using radiation beams to kill cancer cells in human body. In breast cancer cases, radiotherapy could be given as adjuvant therapy following mastectomy, definitive therapy following surgery and palliative therapy on metastatic locations (Williams and Thwaites, 1993).

In order to kill cancer cell in human body, there are different kinds of treatment radiotherapy methods which can be used to deliver radiation energy into human body, namely, radionuclide therapy, brachytherapy, external beam radiation therapy (EBRT) and particle therapy (Metcalfe et al., 2007). However, the most established radionuclide therapies are treatment for thyroid gland cancers, hematologic cancers and bone metastases (Brans et al., 2006). Hence EBRT, brachytherapy and particle therapy are the primary concerns.

This article is intended to assess modern radiotherapy techniques which are clinically used for breast cancer treatment in the world, and find the possibility of the technique to be applied as an option for treating breast cancer in Indonesia.

Conventional Radiotherapy

In conventional radiotherapy, external beam radiotherapy refers to utilization of high energy X-ray or electron beam from a basic linear accelerator or a cobalt-60 beam (Mayles et al., 2007). A pair of beams in the opposed tangential directions is delivered to kill cancer cells in a particular energy range. Since the energy range typically is between 100 kV to 25 MV, it has led to the biological damage of normal tissue adjacent the cancer (IAEA, 2008).

A significant feature from conventional radiotherapy is the application of 2D X-ray films to support the treatment delivery decision. This feature makes it really different from modern radiotherapy, since modern radiotherapy utilize an output from

modern imaging devices such as computed tomography X-ray (CT scan), magnetic resonance imaging, and electronic portal imaging device (EPID) (Thariat et al. 2012). In addition, the transformation from conventional to modern radiotherapy also has been integrated by the development of computer hardware and software (Khan, 2010).

Modern Radiotherapy

Modern radiotherapy makes it possible to create a normal dose distribution of radiation beam during the delivery of radiation dose to the patients (Fraass and Moran, 2012). It ensures that the radiation beams can kill cancer cells, while the normal t issues adjacent the target organs receive a number of radiation doses as low as possible (Levitt et al., 2012). The development of imaging technology has promoted the manufacturers in medical linear accelerators to produce advanced radiation treatment delivery methods. It makes the optimization of radiation patterns in Three Dimensional Conformal Radiation Therapy (3DCRT) give a maximum dose to the target and a minimum dose to surrounding organs (Metcalfe et al., 2007). Moreover, each beam might be assigned by using a three-dimensional computer radiotherapy treatment planning (3DRTP) software to produce a 2D dose display (Khan, 2010). With the assistance of dose display, a computation of 3D dose to PTV can be evaluated by considering the tumor control probability (TCP) and normal tissue complication probability (NTCP). As a result, these planning data are sent to the machine after a verification of patient position, beam placement and dosimetry aspects (IAEA, 2008).

Volume definition in radiotherapy

When a patient has been referred to be treated with radiation therapy, a radiation oncologist will prescribe the treatment dose and consider some volume of tissue adjacent the tumor site. For these purposes, some terms in delineation of the tumor (target organ) and normal tissue have been introduced by the International Commission on Radiation Unit and Measurements (ICRU) in reports 50, where the terms for specific volume targets namely are: the gross tumor volume (GTV), planning target volume

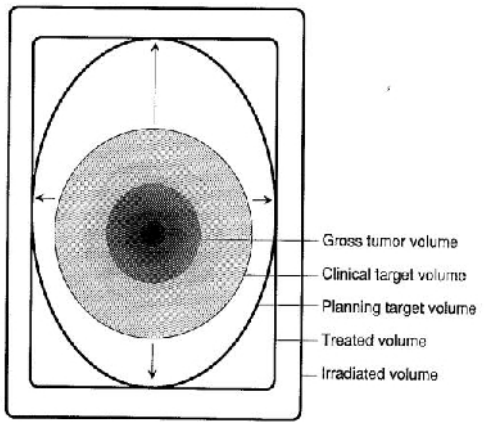

Figure 1. Definition of target volumes as stated in ICRU Report, reprinted from ICRU Report no.50 and 83, respectively (IAEA, 2008).

(PTV), clinical target volume (CTV), target volume and irradiated volume (see Figure 1) (ICRU, 1993).

Recently, the ICRU Report has been updated with the report number 83, which presented new terms in volume definitions, namely organ at risk (OAR), planning organ at risk volume (PRV), internal target volume (ITV), treated volume (TV), remaining volume at risk (RVR) ((ICRU, 2010). The new terms refers to the consideration of delivered dose in by setting up an internal margin for the organ target. For example, an internal margin for breast cancer radiation therapy will consider the variation of the shape, size and position of the tumor with regard to the surrounding organs' such as heart, lungs, muscles, and ribs (Hoskin, 2012).

A qualitative assessment of treatment planning can be done by evaluating the cumulative dose volume histogram (DVH) which displays the volume (or percentage of total organ volume) that receives at least a certain dose threshold (see Figure 2) from reference (Allen et al., 2012). Figure 2.a denotes the DVH for an ideal situation, while Figure 2.b represents the DVH in practice. However, an ideal DVH is difficult to be achieved, since the PTV practically receives less than 100% of the prescribed dose as a result of the PTV overlaps with surrounding organs (Allen et al., 2012).

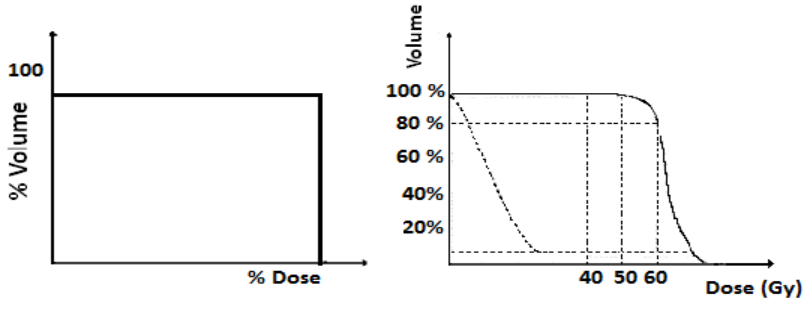

Figure 2. A comparison between an ideal DVH (A) and a practice's DVH (B) (Allen et al., 2012). (Reprint from CSIRO Publication: Biomedical Physics in Radiotherapy for Cancer, permission requested).

Intensity Modulated Radiation Therapy (IMRT)

The utilization of modern multi leaf collimator (MLC) was started in 1980 to replace the conventional collimator. The modern MLC consists of 80 pairs of leaves that have capability to shape modulated beams (Khan, 2010)0. As a result, an advanced method of 3DCRT, called as intensity modulated radiation therapy (Purdy, 2001), has been investigated. Since MLC creates the shaped beams driven by computerized system, it allows the beams move automatically during the beam delivery process. Moreover, the radiation contours can be fitted more closely around the target organ to give a true conformal dose distribution of the PTV. Consequently, for a given radiation dose, the harmful effect to the adjacent organs will be lower (Purdy, 2001; Rudat et al., 2011).

Figure 3 shows a comparison of beam delivery of the IMRT and the 3DCRT. It illustrates that the IMRT has capability to create the dose distribution with a concave shape, despite the time needed to do an iterative optimization might be longer than that of the 3DCRT (ICRU, 1993). This advantage makes the IMRT chosen as the new treatment modality, replacing the 3DCRT to kill localized cancers (Ahamad et al., 2005). Furthermore, a review of clinical applications has shown that the IMRT can reduce toxicity in radiotherapy treatments for breast cancer (Bhide and Nutting, 2010).

Image-Guided Radiation Therapy

With the advantage of high quality of CT or MRI images during treatment planning process, IMRT has achieved better modulated beam to the target organs (Ling et al., 2006). However, during the treatments, there are some important factors which affect the precision of dose delivery, such as interfraction motion within cancer sites, patient motion during the treatment and the deformation of tumor shapes (Herk, 2007). Those factors can play as a source of error such as misalignment and overdose. Hence, in order to control the motion as a source of error, an advanced IMRT has been initiated as the image-guided radiation therapy by utilizing ionizing imaging devices during the treatment process, such as X-ray, and CT scan, as well as non ionizing radiation

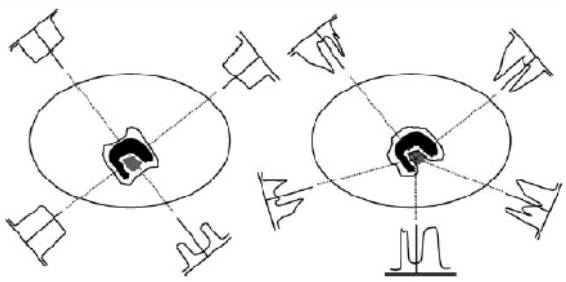

Figure 3. Comparison of beam delivery in 3DCRT (left) and IMRT (right), reprinted from ICRU Report no.83, reference (ICRU, 1993).

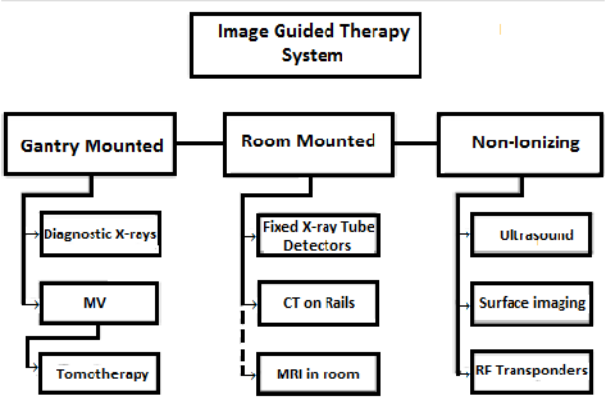

Fig 4. A schematic of combined imaging system in IGRT, the dashed line is shown as the system under development. The horizontal connecting lines indicate possible combinations with different hardware configurations, reprinted from reference (Chen et a l., 2009)0 (permission requested from Springer: Radiological Physics and Technology).

imaging devices such as ultrasound and MRI (see Figure 4) (Chen et al., 2009).

Volumetric Modulated Radiation Therapy (VMAT)

Following the development in radiation therapy methods, Linac vendors have released machines which are able to change the dose rate variation and the speed of gantry. This development leads to the increasing treatment sites that can be treated using external beam radiation therapy, especially in some complicated sites where IMRT cannot be performed effectively (Matuszak, 2010). In 1995, from the reference (in Otto, 2008), Cedric Yu developed Intensity Modulated Arc Therapy (IMAT) to achieve the improvement of flexibility in high conformal radiation. Furthermore, a development called Volumetric Modulated Arc Therapy has overcome some IMAT's limitation by delivering beam in single gantry arcs in 360 degree range of gantry while includes dose rate and gantry speed variation. As a result, the radiation dose is lower, the delivery is shorter, and the total body scatter dose is lower (Walton and Broadbent, 2008). Hence VMAT is a method with combination of effective, flexible and optimized radiation therapy (Ruther, 2000).

Brachytherapy

Brachytherapy technique employs sealed radioactive sources to be placed into adjacent cancer sites. Basically, brachytherapy is not a modern radiotherapy technique since it has been used for more than 100 years (Nicolini et al., 2009). A categorization based on the delivered dose has been acknowledged, namely low dose (LDR) , with a dose rate of less than 2 Gy per hour, high dose rate (HDR), with a dose rate of between 2 and 12 Gy per hour and a medium dose rate (MDR) for the dose between LDR and HDR (Inoue, 2009).

The application of treatment planning software and the modern afterloading system makes the insertion of radioactive become the important technique in modern radiotherapy (Saphiro, 2002), since the digital imaging devices has significant role in defining the accuracy of the insertion of radioactive sources to be placed adjacent the target (Robinson, 2008). There are several radioisotopes which can be used as barchytherapy sources, but only few isotopes are commonly used, such as Co-60, Cs-137, Au-198, Ir-192, I-125, and Pd-103. A summary of physical characteristic of some brachytherapy sources has been listed in Table 1. Hoskin and Coyle (2001) provides the physical characteristic data of common brachytherapy sources. All sources are man-made radiation sources which have been produced using neutron activation in nuclear research reactor.

Brachytherapy can be performed by using an interstitial or intracavitary insertion method, depending on the location tumor sites (Podgorsak and Kainz, 2006). In breast cancer treatment, HDR interstitial brachytherapy may be used temporarily in the location adjacent the cancer (Polgar and Major, 2009) and it could be given after breast conserving surgery (Rulli et al., 2010). A recent technique called as accelerated partial breast irradiation (APBI) has been applied using brachytherapy technique (Lettmaier et al., 2011). Furthermore, a recent study shows that brachytherapy in APBI give less toxicity to adjacent organs and reduced treatment time (Gomeziturriaga, 2008).

Particle therapy

The principle of particle therapy is similar to EBRT. In EBRT, photon is used as the radiation beams, while in particle therapy, the radiation beams are originated from charged particles such as, proton, neutron or light ion therapy (Spoelstra and Senan, 2008). However, at the moment only protons and carbon ions are clinically used, because particle therapy, especially proton therapy, has highly conformal and homogenous radiation beam (MacDonald et al., 2006). These unique characteristics of proton beams are caused by protons depositing their energy until a particular range, called as Bragg peaks, depending on the energy. As shown in Figure 6, a variation of the intensity and the energy of the particles, the modulated Bragg peak can be arranged from the energy of beams. As it is shown, the Bragg peak of protons has been achieved at a depth of ± 150 mm, with the smaller dose is delivered compared to the EBRT beams (10 MV photons) (Levin et al., 2005)0.

Table 1. Physical characteristic of brachytherapy sources.

Isotope	Form	Half life	Average photon energy (MeV)
Co-60	Pellet	5.26 years	1.25
Cs-137	needles, tubes, pellet	30 years	0.66
Au-198	Seeds	2.7 days	0.41
Ir-192	Wire	73.8 days	0.38
I-125	Seeds	60 days	0.028
Pd-103	Seeds	days	0.021

The significant features of proton therapy have been acknowledged to give some disadvantages to pediatric patients. Despite that their critical organs can be avoided to receive the unintended radiation doses, pediatric patients have smaller size body, where some critical organs are really close each others. Moreover, they are still young and their organs are more sensitive to radiation beam (Dinesh Mayani, 2001). On the other hands, since proton therapy also creates internal radiation due to neutron interaction as a result of scattering protons, it can lead to the induction of secondary cancer risk due to low doses to the rest of body. It means that proton therapy may have a higher risk of secondary cancer, more than photon therapy (EBRT) for pediatric patients (Yoon et al., 2010).

In breast cancer, proton therapy has been projected to reduce the dose to heart and lung (Björk-Eriksson and Glimelius, 2005; Dowdell et al., 2008) as it has been reported in a comparative study using Intensity Modulated Proton Therapy (IMPT) and IMRT. Proton therapy is able to give lower dose to organ at risks (OAR) (Weber et al., 2004); Macready, 2012). However, since the operational cost of proton therapy is relatively high, the proton therapy is unlikely to be cost-effective (Anonimous, 2012) radiation therapy (Zelefsky et al., 2012), and needs an analysis of socio-economic cost effective before it has been used as a standard therapy (Spoelstra and Senan, 2008).

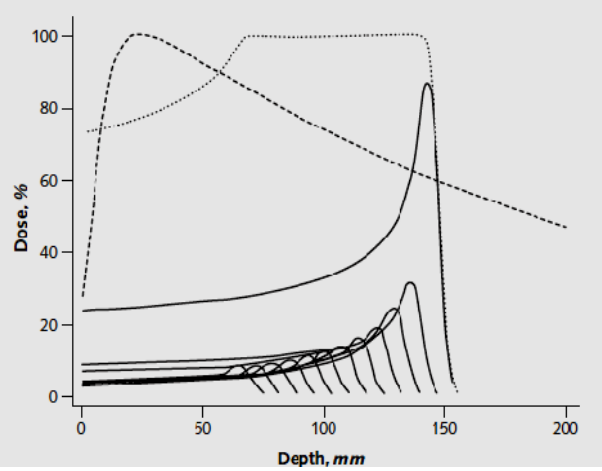

Figure 5. Dose distribution of a spread out Bragg peak of a particle beam compared 10 MV photon beams (Levin, 2005) (adapted with permission from Nature Publishing Group : *British Journal of Cancer*).

Discussion

The main findings show that the updating techniques in modern radiotherapy have attempted to achieve better sparing dose to normal tissues in order to kill cancerous tissue effectively. In order to find the best technique to spare healthy tissue, the references in proton therapy show that proton therapy gives better dose distribution compared to any other radiation therapy techniques and give benefits for breast cancer treatment. In addition, the updating technique such as IMPT also contributes better modulated proton beams in radiotherapy treatment. Unfortunately, most proton therapy facilities are built in high income countries such as US, China, Japan, etc (MacDonald et al., 2006)0.

Furthermore, a recent concern about secondary cancer due to modern radiation therapy has come up during the study (Ruther et al., 2000; Nicolini et al., 2009). Despite that the normal health tissue has been maintained to receive as low as possible dose, but it cannot avoid the probability of secondary cancer due to receiving low radiation dose during radiation treatment. However, a counter argument stated that nevertheless radiotherapy has secondary cancer risk, but the risk can be outweighed by the extension of life expectancy in cancer patients after radiation therapy (Zelefsky et al., 2012).

In addition, it is clear that the beam machines, imaging devices and the treatment planning software are the three important devices in development of modern radiotherapy, but beam machines are the most expensive equipment and play important role in the establishment process of breast cancer management. Furthermore, the more developed technology, the more expensive operation and treatment cost will be. Hence, not all modern radiotherapy can be applied for breast cancer management in low and middle income country like Indonesia.

Due to the lack of academic publication in application of radiotherapy techniques in Indonesia, it is hard to find the current data regarding application of radiotherapy in breast cancer management in Indonesia. As a result a popular article has been found that a hospital (Anonimous, 2012a) has utilized Rapid Arc, a trademark of VMAT from Varian (Anonimous, 2012b), and IMRT (Zelefsky et al., 2012) as standard in radiotherapy. It means that there is possibility to use advanced radiotherapy technology in breast cancer management in Indonesia, and there are a lot of chances to develop the technique not only for breast cancer but also for any other cancer sites. Moreover, these techniques are better and advanced in radiotherapy, but the publication of both methods is still far from number in Indoneisa academic journal. Hence, it needs to be published more in order to share the knowledge and increase public information about the availability of modern radiotherapy in Indonesia.

On the other hand, since the hospital has already utilized IMRT as one of the treatment option in radiotherapy, an IGRT could be developed with the support of any modern imaging devices as listed in the reference (Matuszak et al., 2010), such as CT scan, ultrasound, or diagnostic X-ray. As it has been referred, IGRT is a good radiotherapy technique to improve the motion error during radiotherapy, thus the possibility to improve better treatment outcomes for breast cancer will be possible by adding an IMRT system with modern imaging devices.

The option for VMAT also favorable over IMRT, since VMAT is an improved method from IMRT, some comparison studies also show that VMAT is better than IMRT in delivering conformal beams to some cancer sites, such as breasts (Qiu et al. 2010), head and neck (Hall and Phil, 2006), and prostate cancers (Ruben et al., 2008). In the case of breast cancer, other studies stated that VMAT is useful to reduce the radiation dose of significant organs around the breast such as lung and heart (Popescu et al., 2010).

In addition, brachytherapy has been used for cancer treatment in few hospitals. The treatment cost using brachytherapy eventually is lower than treatment cost using VMAT and IMRT. It means, brachytherapy can be used as one of treatment option in breast cancer.

One of Research Center in National Nuclear Energy Agency (BATAN), namely the Center of Radioisotope and Radiopharmaceutical has conducted research in order establishing brachytherapy seeds for servical cancer, as it has been stated in its Strategic Plans for the 2010- 2014 (PRR-BATAN, 2010). Regarding to this issue, there are many possibilities to develop the technology for breast cancer brachytherapy. Hence, it could lead to the low cost technology for breast cancer treatment without rely on foreign technology resources, which might increase the cost of radiotherapy. As a result, it will give more chance to the breast cancer patients to get access to modern radiotherapy.

Conclusion

To conclude, in regard to application of modern radiotherapy in Indonesia, Intensity Modulated Radiation Therapy (IMRT) and Volumetric Modulated Radiation Therapy (VMAT) can be used as the best option of favorable technique for breast cancer due to its abillity to reduce the radiation dose delivered to the organs at risk. Both techniques have been used in few hospitals in Indonesia. With these techniques, there will be possibility to develop technique to gain better treatment outcome in breast cancer. In addition, we are looking forward an established brachytherapy technology from the Center of Radioisotope and Radiopharmaceutical, with Ir-192 as one of options in breast cancer management in Indonesia.

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Radiation-induced DNA Double Strand Breaks and Their Modulations by Herbal Treatments: A Cancer Cell Culture Model *)

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Abstract. **Gamma radiation brings deleterious effects upon human cells by inducing oxidative stress and damages. Antioxidants have been shown to confer protective effects on irradiated normal cells. *Moringa oleifera Lam*. is a widely used nutritional supplement with antioxidant activities. This report showed that antioxidant-containing supplements, in addition to protecting normal cells, could protect cancer cells against genotoxic effects of gamma radiation. γ-H2AX immunofluorescent foci were utilized as an indicator of radiation-induced DNA double strand breaks. MCF-7 human breast adenocarcinoma cells were irradiated with 2-8 Gy gamma radiation. A linear relationship between the formation of γ-H2AX foci and radiation dose was observed with an average of 10 foci per cell per Gy. A 30-minute pretreatment of the cells with either the aqueous or the ethanolic extract of *M. oleifera* leaves could partially protect the cells from radiation-induced DNA double strand breaks. A pretreatment with 500 µg/mL aqueous extract reduced the number of foci formed by 15% when assayed at 30 minutes post-irradiation. The ethanolic extract was more effective; 500 µg/mL of its concentration reduced the number of foci among irradiated cells by 30%. Our results indicated that irradiated cancer cells responded similarly to nutritional supplements containing antioxidants as irradiated normal cells. These natural antioxidants could confer protective effects upon cancer cells against gamma radiation.**

Keywords: gamma radiation, MCF-7, DNA double strand breaks, γ-H2AX, antioxidant, Moringa oleifera

*) The complete paper will be published in Atom Indonesia Journal

In Vitro Study on the Immunogenecity of Irradiated Sporozoites of *Plasmodium falciparum* in HepG2 Cell Lines

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Abstract. **Malaria is initiated by *Plasmodium* sporozoites infecting the liver. Preventing this infection would block the obligatory first step of infection and perhaps reduce disease severity. This prevention relies primarily on cytotoxic lymphocyte activity against infected hepatocytes. However knowledge on the suppression by immune response induced by irradiated sporozoites is still limited. Aim of this study was to explore the immune response elicited in hepatocytes cell after irradiated sporozoites in vitro inoculation. Sporozoites obtained from infective *Anopheles sp.* were in vivo irradiated with gamma rays at doses of 150, 175 and 200 Gy and then were inoculated into in vitro culture of HepG2 cell lines and at 1, 3 and 20 hours the CD8 contents were measured with ELISA. Results showed that there was a response in term of CD8 concentration post inoculation of irradiated sporozoites in cell line. The level of $CD8^+$ T cell activation varied with the length of time of hepatocyte infection. The high immune response mainly was seen at dose of 175 Gy of gamma rays and within 3 hours inoculation. These results lead to the presumption that the sustainable protection against malaria associates with distinct intra-hepatic immune responses characterized by strong interferon-γ producing CD8+ memory T cells.**

Keywords : malaria vaccine, sporozoites, immune response, CD8 T cell, HepG2

Introduction

Malaria is a potentially life threatening sickness caused by the single cell parasite *Plasmodium* (McCarthy, 2003). This infection disease is initiated by *Plasmodium* sporozoites invading the liver. Preventing sporozoite infection by vaccination would block the obligatory first step of the infection and perhaps reduce disease severity (Frevert et al., 2005). Among the various strategies proposed for antimalarial vaccination, live-attenuated *Plasmodium* parasites, such as radiation-attenuated plasmodia sporozoites (RAS-spz), remain the gold standard because they confer long-lasting protection against natural malaria transmission in malaria-naive humans and laboratory rodents (Hoffman, 2002; Jobe et al., 2007). Irradiated sporozoites infect hepatocytes *in vivo*, as normal sporozoites do; however, irradiated sporozoites do not progress further to blood-stage infections and, therefore, do not induce malaria-associated pathology (Good et al., 1988; Silvie et al., 2002; Suhrbier et al., 1990).

Protection induced by vaccination with RAS-spz is essentially mediated by interferon (IFN)-γ-producing CD8+ T cells (Good et al., 1988; Krzych et al., 2000; Guebre-Xabier et al., 1999; Doolan et al., 2000). Antibodies generated in response to attenuated parasite vaccines also contribute to protection, but CD8+ T cells are believed to play the major protective role (Doolan et al., 2000; Rodrigues et al., 1993). CD8 T cells appear to be the *sine qua non* effectors that confer sterile immunity in some strains of attenuated sporozoite-immunized mice (Tarun et al., 2007). CD8 T cells specific for Pf circum sporozoite (CS) protein have also been found in the blood of both RAS-immunized volunteers and naturally exposed subjects (Doolan *et al*., 1997; Malik et al., 1991). There is evidence that some sporozoites migrate from the dermis to the draining lymph node where they activate CS-protein-specific CD8 T cells (Chakravarty et al., 2007). These CD8 T cells subsequently travel to the liver where they could contribute to pre-erythrocytic stage immunity. It is more likely, however, that attenuated sporozoites also induce CD8 T cells that are specific for the liver-stage Ags developing only in infected hepatocytes. Attenuated sporozoites activate CD8 T cells, whereas native sporozoites do not, and this could be related to the former causing infected hepatocytes to undergo apoptosis (James, 2005; van Dijk et al., 2005).

This research aimed to understand more fully the basis of the immunity generated by the exo-erythrocytic parasites derived from RAS-spz at doses and dose rate of irradiation that are quite different with other research, their antigenic repertoire was studied in an *in vitro* system with the human parasite *Plasmodium falciparum* and the human hepatoma line HepG2 as host liver cells.

Materials and Methods

P. falciparum in vitro culture

Asexual stages of *P. falciparum* 3D7 strain were maintained *in vitro* in 5% hematocrit in RPMI 1640 medium containing 0.5% albumax and 80 mg/ml gentamycin sulphate in a humidified chamber containing 5% CO_2 at 37°C in human erythrocytes as described earlier (Dalton et al., 1993). After every 24 h the media was removed using a sterile Pasteur pipette without disturbing the cells that settled down. Then the cells were mixed without frothing and a drop of blood was placed on the slide and a thin film was made. Fresh complete media (with 10 % serum) was added, mixed properly, and kept back in the incubator. Monitoring of parasitemia is accomplished by preparing blood films, staining with Giemsa stain following methanol fixation, and counting infected red blood cells microscopically.

Mosquito infections with membrane feeding

Anopheles sp. were reared in controlled environments (27°C and 80% relative humidity) (Doolan et al., 2000). Membrane feeding was performed using *P. falciparum* culture mixed with fresh venous blood. Standard medium-sized membrane feeders were used, that used circulating water to maintain a temperature of 37°C. For each dose of irradiation, a cup of 80 mosquitoes was placed in the feeder, and mosquitoes were allowed to feed for 90–120 min (Diallo et al., 2008). The engorged mosquitoes were taken and kept in a carton special cup covered with filter and placed in transparent plastic at 21^{o}C temperature controlled room. To confirm infection, oocyst prevalence were monitored 7–8 days after infection by taking 2-3 mosquitoes from the batch and other mosquitoes were maintained in conditioned room for 14-15 days.

Irradiation of mosquitoes

After 14-15 days, mosquitoes infected with *P. falciparum* sporozoites were *in vivo* exposed to 0, 150, 175, 200 Gy dose of γ-radiation using a ^{60}Co source. The time of exposure of infected mosquitoes to achieve the target radiation doses was based on the calibration of the irradiator by dosimetry and the half-life of ^{60}Co.

Preparation of irradiated sporozoites

Salivary glands containing sporozoites were isolated from batch of *P. falciparum* infected mosquitoes using both manual isolation and a modified microcentrifugation technique described previously (Ozaki et al., 1984). Briefly, for manual isolation, mosquitoes were anesthesized in freezer for 3 minutes, their wings and legs were removed and placed on a glass slide, and salivary glands were placed in a 1.5 ml microcentrifuge containing 250 µl of physiological solution. For centrifugation isolation, head and thoraxes of mosquitoes were separated, and the abdomens discarded. Up to 10 heads and thoraxes were placed on a 0.5 ml microcentrifuge tube previously perforated with a 25-gauge needle and plugged with glass wool. This tube was placed in a 1.5-ml microcentrifuge tube, and 50 µl of buffer (PBS, pH 8, 1.5% glucose, 5% fetal bovine serum) was added and the tubes were centrifuged for 2 min at 10,000 × *g* in an Eppendorf microcentrifuge. Filtrates were collected, and centrifugation was repeated after the addition of another 50 µl of buffer.

Infection of HepG2 cells in vitro

HepG2 human liver cells were maintained in complete Dulbecco's modified Eagle's medium (DMEM) buffered with bicarbonate and supplemented with 10% fetal bovine serum at 37 °C in a 5% CO_2 environment (Bai et al., 2001). For infection by *P. falciparum* sporozoites, 1×10^6 cells were cultured in DMEM in 6-well plates at 80% confluence and exposed to freshly prepared sporozoites. Cells were exposed to sporozoites for different time periods (1, 3, and 20 h), after which the sporozoite-containing medium was aspirated from each well for CD8 concentration assay.

CD8 cellular determination with ELISA

The antibody test is based on binding of anti-*Plasmodium* antibodies present in sample to antigens immobilized on 24-well plates (Bonelo et al., 2000). The test was done as recommended by the manufacturer, as follows. Ready-to-use diluent buffer (125 µL) was dispensed into each well, followed by 25 µL of test serum. On the same plate, 25 µL positive control and negative control were also dispensed in single well and triplicate wells respectively. The plate was then covered and incubated for 60 minutes at 37°C before being washed 5 times. Horseradish peroxidase-conjugated rabbit antihuman IgM and IgG monoclonal antibodies (100 µL) were added to each well and the plates were incubated for 30 minutes at 37°C. The wells were again washed 5 times, and 100 µL of substrate solution were added to each well. The plate was covered and incubated in the dark for 15 minutes at 37°C. Finally, 50 µL of 0.5 M sulphuric acid was added to each well and absorbance was read within 15 minutes at 450 nm, with a reference wave-length of 620 nm. The standard solution was used for all experiments.

Results and Discussion

In vitro data of this experiment showed that primary hepatocytes are capable of processing and presenting sporozoite antigens after infection (Figure 1). This result was consistent with other experiments (Schofield et al., 1987; Romero et al., 1989; Guillouzo, 1998). However, the level of CD8 T cell activation

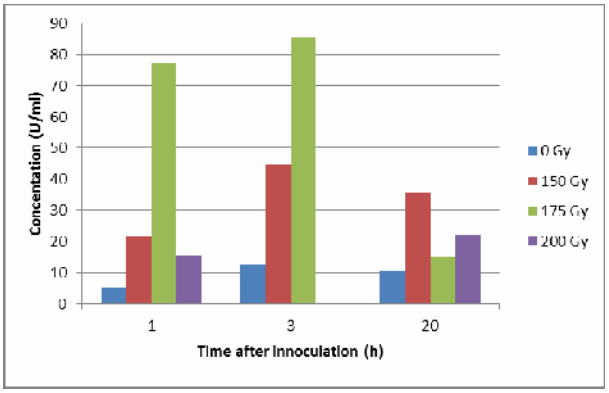

Figure 1. The concentrations of CD8 T cell (U/ml) in HepG2 culture medium at various times after inoculation of irradiated sporozoites.

varied with the length of time of hepatocyte infection. To find out the optimal time necessary for processing after infection with sporozoites, hepatocytes were infected with sporozoites for different time periods over a total period of 20 h. Concentration of CD8 T cells after coculture with infected hepatocytes was determined by an ELISPOT assay. Optimal activation of the T cells occurred when hepatocytes were infected with the sporozoites for a time period between 1 and 20 h. Activation was significantly high when T cells were cocultured with hepatocytes infected with sporozoites for 3 h (Figure 1). This suggests that the optimal time necessary for hepatocytes to process to CD8 T cells after infection with sporozoites is between 1 and 3 h. Data of evaluation showed that immune responses based on the concentration of CD8 can be induced in *in vitro* cell culture of HepG2 cell lines, mainly at dose of 175 Gy of gamma rays and within 3 hours inoculation.

Taking into consideration that T cells could be activated by contaminants of parasite preparations,

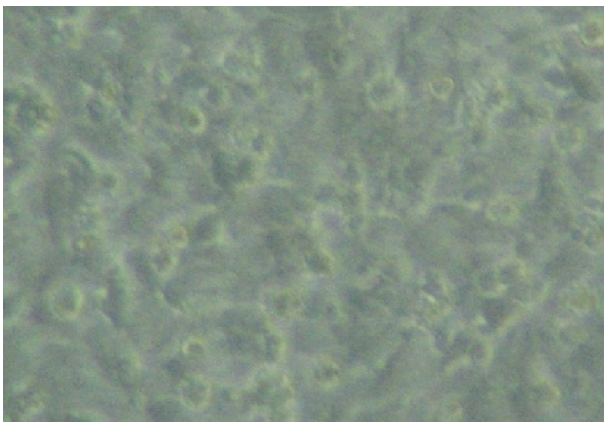

Figure 2. Microscopic view of *in vitro* culture of HepG2 cell lines.

hepatocytes were incubated with material from uninfected mosquito salivary glands as a control. The activation of T cells observed is specific for sporozoites, because T cells were not activated in the presence of uninfected salivary gland material. To eliminate the possibility of Ag from extracellular degradation being presented, hepatocytes were incubated with heat-killed and with midgut sporozoites (which contain the mature CSP, but neither migrate through nor infect hepatocytes). T cells were not activated in either case, implying that migration and/or infection of sporozoites are necessary for the CSP to be processed and presented by hepatocytes.

As found in this experiment, irradiated sporozoite-induced protective immunity has been shown to involve $CD8^+$ T cells directed against liver stage parasites, and in vivo depletion of this effector has led to blood infection upon a live sporozoite challenge (Schofield et al., 1987; Romero et al., 1989).

Cultures of primary hepatocytes and hepatoma cell line HepG2 are frequently used in *in vitro* models for human malariology studies. Primary hepatocytes provide the closest in vitro model to human liver. Furthermore, evaluation of human hepatocytes as the irradiated sporozoites efficacy analysis of erroneous predictions should facilitate further refinements to malaria vaccine development strategies. *In vitro* liver preparations are increasingly used for the study of attenuated parasites. In recent years their actual advantages and limitations have been better defined. The primary hepatocyte cultures, appear to be the most powerful in vitro systems, as liver-specific functions and responsiveness to inducers are retained either for a few days or several weeks depending on culture conditions (Guillouzo, 1998).

Primary human hepatocytes and hepatoma cell lines such as HepG2 are among the most widely used in vitro models in pharmacological and toxicological studies. Primary human hepatocytes remain differentiated and sustain the major drug-metabolizing enzyme activities for a relatively long period of time in culture; they represent a unique *in vitro* system and serve as a "gold standard" for studies of drug metabolism and toxicity (LeCluyse, 2001). On the other hand, HepG2 hepatoma cells are relatively easy to maintain in culture and are widely used for toxicity studies.

A substantial and protective response against malaria liver stage involves induction of CD8-T cells. From experiment by Bongfen et al. (2007), it can be hypothesized that infected hepatocytes can contribute in vivo to the elicitation and expansion of a T cell response. Other research by Van Dijk et al. indicated that immunization of mice with *P. berghei p52*$^-$GAP results in immune responses that were comparable to

those induced by RAS or GAP lacking expression of UIS3 or UIS4, with an important role implicated for intrahepatic effector memory $CD8^+$ T cells.

An effective malaria vaccine is needed to address the public health tragedy resulting from the high levels of morbidity and mortality caused by *Plasmodium* parasites. The first protective immune mechanism identified in the irradiated sporozoite vaccine, the "gold standard" for malaria preerythrocytic vaccines, was sporozoite-neutralizing antibody specific for the repeat region of the surface circumsporozoite (CS) protein. The obstacles in the creating malaria vaccine are the finding of vaccine material that is specific the host to be immunized and models used in pre-clinical test. An effective malaria vaccine should also induce strong and long-lasting pre-erythrocytic stage T cell immunity. The identification of the most protective pre-erythrocytic stage Ags is a primary objective.

Bongfen et al. (2007) investigated how primary hepatocytes from BALB/c mice process the CSP of *Plasmodium berghei* after live sporozoite infection and present CSP-derived peptides to specific H-2Kd-restricted CD8-T cells in vitro. Using both wild-type and *spect(-/-) P. berghei* sporozoites, they showed that both infected and traversed primary hepatocytes process and present the CSP. The processing and presentation pathway was found to involve the proteasome, Ag transport through a postendoplasmic reticulum compartment, and aspartic proteases. Thus, it can be hypothesized that infected hepatocytes can contribute in vivo to the elicitation and expansion of a T cell response.

P. falciparum sporozoites, as found in this research and many other findings in laboratory, confer protection when irradiated at 12–15 krad (120-150 Gy), but not at higher doses. It is generally admitted that a moderate but protective irradiation dose preserves the capacity of *P. falciparum* sporozoites to invade hepatocytes and transform into blocked uninucleate trophozoites in the liver, while higher and non-protective doses are believed to limit the sporozoite efficiency to invade liver cells (Mellouk et al., 1990).

In this research a dose of several hundred grays (150-200 Gy) is required to attenuate, in the reproductive sense, the microorganism cells whereas the lethal dose for mammalian cells is only a few grays. The mechanisms responsible for these differences are not well understood, but could involve more efficient mechanisms of DNA repair of the damage caused by gamma rays. The inability of 6.5 kGy-irradiated cells to divide results from unrepaired double DNA breaks, leading to a cell cycle arrest or unbalanced chromatin exchange in daughter cells and consequent mitotic death (Rhind et al., 1998; Frankenberg, 1998). By contrast, over-irradiation possibly prevents the expression of genes and thus the neosynthesis of liver-stage proteins. If too little radiation was given to the mosquitoes the parasites weren't weakened enough and you actually infected the people when you exposed them to the mosquitoes. And if you gave too much of radiation the parasites became too weak and they didn't illicit a strong enough immune response to protect against malaria (Hoffman, 2002).

Several problems were faced in this vaccine material development research. Sporozoite suspensions used for immunization may heavily contaminated with microorganisms and mosquito components. An alternate approach, however, allowed irradiated mosquitoes to directly inoculate attenuated sporozoites into hosts, the mosquitoes thereby acting as vehicles of immunization. Moreover, the use of radiation for sporozoite attenuation is not without risk, because it yields heterogeneously attenuated sporozoite. This process is also radiation dose sensitive (Mellouk et al., 1990), and under-irradiated spz remain infectious, whereas overirradiated spz are not sufficiently immunogenic to prevent infection (Mellouk et al., 1990; Silvie et al., 2002). These problems prompted a search for other forms of attenuation that would render the parasite a more reliable vaccine. Thus, current technology limits the feasibility of their commercial scale production and poses a significant risk of contamination with transmissible agents.

Conclusion

Experiment showed that there was a response in term of CD8 concentration post inoculation of irradiated sporozoites in cell line. The level of $CD8^+$ T cell activation varied with the length of time of hepatocyte infection. The high immune response mainly was seen at dose of 175 Gy of gamma rays and within 3 hours inoculation. Our data show that sustainable protection against malaria associates with distinct intra-hepatic immune responses characterized by strong interferon-γ producing CD8+ memory T cells.

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apoptosis of infected liver cells. *Proc. Natl. Acad. Sci. U.S.A.*, 102, 12194–12199.

DISCUSSION

Question :

Dr. Ben Suprayogi, PORI:

You got the effective dose of gamma rays to attenuate parasites was 175 Gy, what was the implication of this finding, and also you got the time of highest CD concentration was 20 hours, what was the implication of this?

Answer:

This implicate that 175 Gy is the best or the most effective dose used to create a vaccine materials for malaria, and 3 hours implicate that the immune response of liver cells is initiated within a quite short time (3 hours).

Question :

Dr. Iwan, PORI :

The dose used to irradiate paratites is 150 Gy or 150 centiGray?

Answer:

The dose used is 150 Gray, not centigray, and this is a very high dose required to attenuate parasites which has a small number of chromosome (11 pieces) in its nucleus and also the parasites is exist in the red blood cells which has a sitoplasm as a barrier.

Roots as Medium for Measuring TENORM Radiation Around the Gas Industry Area

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Abstract: **A study concerning the presence of TENORM in the gas industry by using kale and water hyacinth as samples have been done. The previous study informed that contaminated kale has a very high radiation activity. Current study had been conducted to measure the radiation contaminated TENORM areas in the gas industry. Measurements were performed using kale and water hyacinth plants that grow in that environment. This study only measured the radiation from the soil that detected from the kale and water hyacinth plants grown and also at the root. The sample used in this study was the kale and water hyacinth soil and waste (liquid) taken from landfill gas plants in one industry in North Aceh. Sampling site was divided into two regions, namely region I and region II and both areas are in the category of regional gas industry with 14 samples. At region I, it was observed that the calculated value of the radiation in soil was increased about 10% compared to background values and the lowest increasing was about 5%. At region II, the highest value in root was increased about 28% compared to soil value, and the lowest value was about 2% compared with background value. The Kale and water hyacinth plants that grown in the gas industry can be used as good indicators to detect the level of radiation exposure to TENORM.**

Keywords: radiation, contamination, industry gas environmental.

Introduction

TENORM (technologically enhanced naturally occurring radioactive material) is the natural radioactivity due to human activity or process technology has increased the potential radiation exposure, compared to the initial state. TENORM focuses on the results of the waste industry. TENORM, in accordance with its meaning, is a radioactive material derived from nature (rocks, soil, and minerals) and concentrated or increased content of radioactivity as a result of industrial activity. TENORM is found in uranium mining, phosphate fertilizer production plants, oil and gas, geothermal energy production (Akhadi, 2009).

The preliminary study to observe the presence of TENORM in small area of industrial environment by using phytoremidiation technique which uses kale plants was done in similar region. The result shows that the count of percentage of samples that grow within this area exceed 100% from other samples for measurements during 3600 s. It can be informed that the kale was contaminated and it has very high activity (Safitri, et al, 2012). This result is appropriate and supports the results of some plants that have been test its ability to absorb various radioactive substances such as water hyacinth and India lettuce plants (Setiawati, 2004). The ability of organisms to accumulate radioactive materials so that the concentration in biota, is very high above the concentration of media that is a pathway of radioactive materials used for the study biological indicators in case of radioactive contamination in the environment. Based on these conditions, the fieldwork has been done to measure the area of operation of TENORM radiation contamination gas industry using kale plants and water hyacinth. This study only measured the radiation from the soil where the chopped kale and water hyacinth grows and the two roots of the plants.

The presence of TENORM in the mining/industry area can be formed as radioactive materials that concentrated through the production process. Scales that were attached at the pipe (magnification of salt on the surface of oil and gas pipelines), residues and sludge is an object or a place where TENORM can be found. TENORM can be either raw materials or by-products (by product), either in the form of gas, solid (scale and slag), mud, a thin layer (film) or liquid TENORM is a natural radioactive substances due to human activity or technology process have increased the potential radiation exposure, compared to the initial state that focuses on industrial process wastewater results. Radioactive substance is radioactive material in the form of a physical nature. TENORM, in accordance with its meaning, is a radioactive material derived from nature (rocks, soil, and minerals) and concentrated or increased content of radioactivity as a result of industrial activity. TENORM is found in uranium mining, phosphate fertilizer production plants, oil and gas, geothermal energy production (Kosako et al., 2005).

Materials and Methods

The samples that used in this study were kale, water hyacinth soil and waste (liquid) taken from a landfill of gas plant at North Aceh. Sampling site was divided into two regions, namely region I and region II.

Both areas were in the category of gas industry region. There were 14 samples taken from both regions. Samples of soil and roots shaped its kale plants or water hyacinth. Then the samples were separated in the container, weighed of each sample of 20 grams. Samples were cut to get the roots, rootstock, the stems and leaves. The next section was weighed and put into containers. Sample enumeration was performed using scintillation detector. Enumeration process for each sample was 1 hour. The sample measurement scheme can be seen in Figure 1, and consisting of the container for placing the sample; scintillation detector; census taker; stop watch; and fertilization and incubation.

Figure 1. Set up experiment Style and Format.

Results and Discussion

Results of previous studies showed that the roots of plants that grow in regional industrial environment contains radioactive material. Radioactive material was thought to originate from TENORM (Safitri, 2012). Based on the results of this study, it was developed more advanced study. Radioactive measurements consist of 14 samples were divided into region I and region II. Each region is represented in 7 samples. In region I, it is observed that radiation count values in the root is reach 10% is in increase in digital background, and the lowest 5% compare the background.

Measurement value in the soil at the lowest price 10% and 0%. It is understood as that the first measurement area with plants and soil samples concluded that the soil is contaminated but still within tolerable limits (Figure 2).

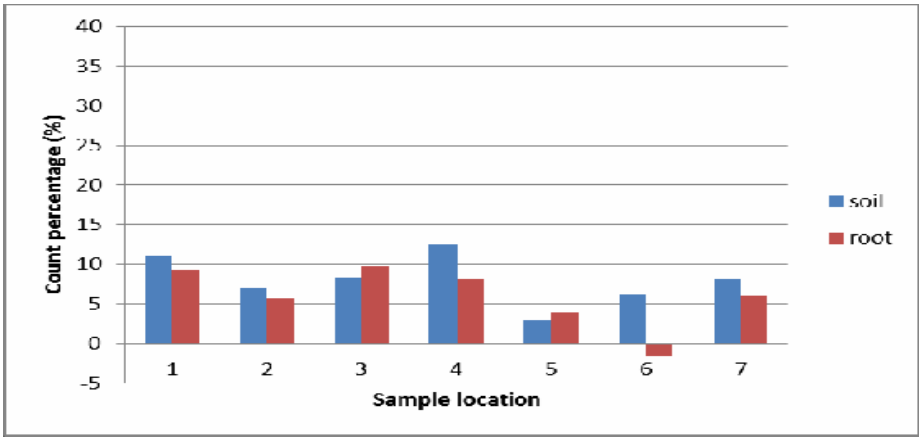

Figure 2. Count percentage of radiation at region I.

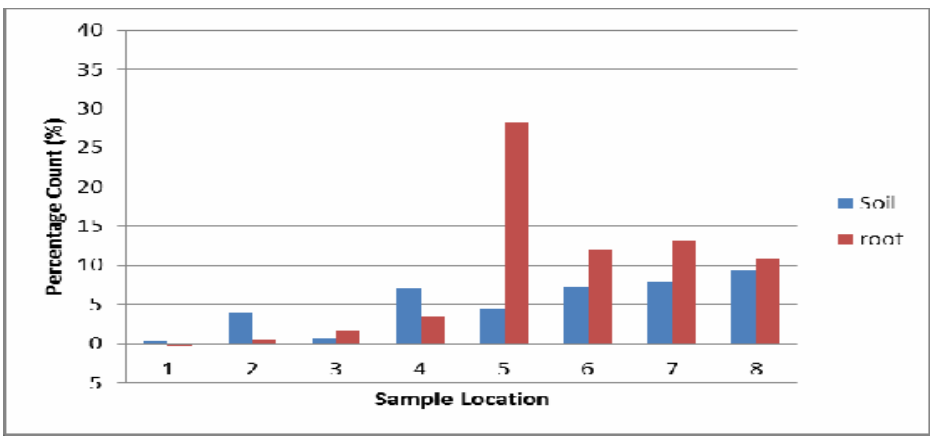

Figure 3. Count percentage of radiation at region II.

For measurements in the region II the highest count values in the soil increased 10 % and the lowest 2% compared to background values. Enumeration of radiation on plant roots increased 28% from the value of the background and the lowest 2%. From these data it can be concluded that the plants were taken in an industrial environment have actually been contaminated with radioactive material source. The measurement results are also supported by the Gamma-ray measurement data in the gas industry (Figure 3). This measurement is the result of cooperation with PTKMR BATAN. The average value for the dose rate measurements in the area of North Aceh the lowest score and 0,047 µSv/hr the highest value 0,069 µSv/hr (Table 1).

Table 1. Dose rate of the samples from any locations.

No.	Location	Dose rate (µSv/hr)
1	Jalan Raya Sampoinit	0,061-0,069
2	Pengadilan Negeri Lhoksukon	0,056-0,068
3	Lhoksukon	0,065-0,064
4	Masjid Gedong Lhoksukon	0,059-0,064
5	Kandang	0.051-0.059
6	Meunasah Gp Batuphat Timur	0,047-0,051
7	Krueng Gekueh	0,057-0,062
8	Kruung Mane	0,051-0,059
9	SMAN Peusangan Mt Glp 2	0,057-0,063

Radiation measurement samples obtained from 14 locations and gamma radiation measurements carried out simultaneously. This study can be observed that the surrounding gas industry, if we doubt the nature of the source of radioactive contamination we can analyze by taking plants growing in an environment in which plants grow. Plant that has been chosen is kale or water hyacinth plants. Both of these plants are very easy to grow plants around the gas industry. Moreover, both plants have proven to be very effective binding of radioactive elements contained in the soil. This study attempts to enrich the research support TENORM radiation in the environment.

The results of gamma-ray radiation exposure, that measured using gamma detectors in several cities in North Aceh shows in Table 1. The value of exposure ranged from a low of 0.047 µSv/hr and the highest 0.069 µSv/hr. Level of exposure depends on the location. The measurement result is smaller than the measured value of radiation exposure at the gas industry environment (Table 2). In addition, the results of the industry survey is not yet complete and the above figures are under acquisition of the mining industry survey. TENORM in mud contaminated water is almost equal to the scale. The rest occurs in the form of oil and mud sometimes containing silica and barium compounds (Rusdiyan, et al. 2006).

The presence of TENORM in the gas industry has been observed by using phytoremediation kale plants that growing around the area. This study research is very important to the community who live around the gas industry. As is known that post-operation of the industry, it would produces industrial waste and it would influences the environmental and caused a significant result of highly radiation exposure to environment. In fact, in the former industrial area is now being used by local people for farming. The measurement technique of radiation exposure by using the plant as object is very necessary to developed. In addition, the costs used are also not too expensive.

Some preliminary studies have been conducted to determine the negative impact of an industry after its operation, where some of the conclusions that can be drawn from these studies are environmental and radiological monitoring program. It is part of the license's conditions and should be carried out in proper manner as the monitoring result. This will represent that TENORM activities to be implemented by the licensee is safe and poses no radiological hazard to the community and does not contaminate the environment. So that, there are a few of license's conditions that need to be reviewed accordingly based on the processing activity (Lin et al., 2011).

The main sources of the contamination are the radionuclides from the U-238 and Th-232 series, which are present in the geological formations that contain oil, gas and water. When water is brought to the surface along with oil and gas, radium, which is dissolved in the produced water, co-precipitates with barium, strontium, or calcium sulfates thus forming scale and sludge. The main radionuclides of concern from the radiological point of view are Ra-226 and Ra-228. The scales and sludge of industrial disposal can contain high quantities of radium and other decay products that can cause exposure of maintenance and other personnel to hazardous radionuclides concentrations.

Table 2. Summary of the survey results (identification) NORM in the oil and gas industry mines.

Mine	Radionuclide	Activity Bq/g	µR/hr	Location	Expanation
Oil and Gas	Ra-226 Ra-228 Th-228 K-40	0,009 ~ 75,376 0,021 ~ 76,246 0,001 ~ 48,811 0,011	10~ 2200	Tanks, pipes, and materials former process	Max value for scale in pipe former

Conclusions

The Kale and water hyacinth plants that grow in the gas industry can be good indicator to detect the level of radiation exposure to TENORM to enumerate the roots of plants. This method may be an interesting and creative simple method to determine the radiation exposure in the environment. The results of this research that was conducted in two region of the gas industry shows that if soil containing radiation sources then it will detect the radiation as well as in the roots of plants contaminated with radioactive sources.

Acknowledgements

This research was carried out using BOPTN Research Grant. We would like to thank the student and laboratory staff of Department of Physics, Faculty of Sciences, Syiah Kuala University in Aceh Province who participate in this study.

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DISCUSSION

Kanokporn B. - Thailand

Questions :

1. What was the minimum samples you toke to get a significant results in radioactivity measurement?.
2. What was the method/tool used to measure that radioactivity?

Answers:

1. The minimum sample was 20 gram
2. The method was that the samples were cut to get roots and the next section was weighted the samples and put into container.

Togap Marpaung - BAPETEN

Question :

What do you think about TENORM from oil and gas mining? Is it a radioactive waste or not. What is your recommendation to the owner of oil and gas mining regarding about TENORM which containing radium with high level activity?

Answer :

TENORM from oil and gas mining is radioactive waste from technology process, radioactive material derived from nature (rocks, soil and mineral). I did not make any recommendation yet but I suppose the government have attention for the TENORM and the environmental.

Interaction of ^{137}Cs and ^{90}Sr With Soil from Rembang and Sumedang as Hostrock Candidate of Radwaste Disposal Facility *)

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Abstract: **Introduction of Indonesian first nuclear power plant (NPP) at Jawa Island will contribute a huge of radwaste volume and therefore a radwaste disposal facility in Jawa Island should be initiated. Rembang and Sumedang areas had potential host rock for disposal facilities due to the nature of rock hydraulic conductivity which is relatively low (~ 10^{-7} m / s) to prevent the possibility of radionuclides contamination spreading into environment. To study the reliability of soil from Rembang and Sumedang area a series of sorption experiments such as contact time, variation of ionic strength and CsCl - $SrCl_2$ in the solution were performed in batch method. Radiocesium and ^{90}Sr had been used as models of metal ion due to dominantly radionuclide in the inventory of low-medium level radioactive waste, and as a reference radionuclide in the study of radionuclide interaction with soil rock or mineral. Objective of the experiment was to find out specific data of radionuclide sorption characteristic of Rembang and Sumedang's soil. Soil and radionuclide was contacted with shaking method, solution were checked periodically to obtain sorption kinetic, the effects of CsCl and $SrCl_2$ concentrations and ionic strength in solution. The specific data could be obtained by comparing the concentration of metal ion at the initial and final activities. Results showed that sorption equilibrium of ^{137}Cs and ^{90}Sr into soil was reached after 5 days contacted with Kd value around 3300-4200 ml/g where Kd factor of ^{90}Sr was higher than for ^{137}Cs. Increasing the concentration of CsCl and $SrCl_2$ in the solution had reduced the Kd value of ^{137}Cs and ^{90}Sr by soil, and the presence of NaCl as background salt in the solution also affected Kd value due to competition among metal ions into soil samples. This information is expected could provide an important input for the planning and design of radioactive waste disposal system in Jawa Island in the future.**

Keywords: Radwaste disposal, host rock, Rembang and Sumedang soil, ^{137}Cs and ^{90}Sr.

*) The complete paper will be published in Atom Indonesia Journal

In vivo Radioadaptive Response of White Blood Cells (*Leukocytes*) Quantity in Mouse (*Mus musculus L*) Blood to Co-60 Gamma Radiation Exposures

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Abstract. **Low radiation dose induction might changes the mechanism of cellular and molecular systems, with a certain condition; it can protect cells to reduce the effect from subsequent high dose. This phenomenon is called radioadaptive response. In this work radioadaptive response has been investigated in 80 male mice with the age of 37 to 46 days old, and the body weight of 23.79 to 26.66 grams. These mice were divided into 4 groups, one group was as control, and the other groups were treated with Co-60 gamma radiation which called as treatment without adaptive, adaptive I, and adaptive II responses. To the mice from the group of treatment without adaptation, challenge doses of 1.0, 1.5, 2.0, 2.5, and 3.0 Gy were delivered. The same challenge doses were given to the adaptive I and adaptive II treatment group directly and 5 minutes after receiving the adaptive dose of 0.1 Gy. The number of total leukocyte counts of the control group was (9.51 ± 0.81) x 103/µl. In general radiation doses decrease the leukocytes counts from 3 treated groups and linearly related with the increasing dose. The effect of radioadaptive response of the adaptive II treatment group was relatively higher, which were illustrated by the linear coefficient of the group without adaptive, adaptive I, and adaptive II with the value of -0.18, -0.15, and -0.11, respectively. These results indicated that the adaptive response increased when there was interval delivering time between adaptive and challenge dose.**

Key words: Induction of radiation, adaptive response, the dose challenges.

Introduction

New mechanistic cell and molecular studies on the effects of very low doses of radiation have resulted in three major paradigm shifts. These phenomena are the big shift in the thinking about the alteration that exists in biologic material caused by ionizing radiation. The three phenomena are adaptive response, bystander effects, and genomic instability which are the important part in molecular, cellular and body tissues response to ionizing radiation (Brooks, 2005).

The first paradigm shift is the alteration from assumption that to induce responses or effects radiation should interact directly with cells and transfer its energy to these cells. In fact cells that indirectly exposed to radiation (unirradiated) may give a response as directly exposed cells. The biological effects that occurred in indirectly exposed to radiation but it exist nearly to exposed cells called as bystander effects (Alatas, 2004) that is defined as the occurrence of biological effects in unirradiated cells as a result of exposure of other cells in the population to radiation.

The second paradigm is the genomic instability which has an important role in cancer induction. Previously, it was thinking that DNA damage occured directly by low dose radiation. Currently it is known that various alterations may expressed several generations after exposure of the original progenitor cell to radiation. Genomic instability in cells can be induced by radiation, that may alter genomic stability of normal cell to be unstable. Genomic instability is a term used to describe a phenomenon that results in the accumulation of multiple changes required to convert a stable genome of a normal cell to an unstable genome characteristic of a tumor. The ability to alter radiation response by physical and chemical treatments suggests that it may be possible to intervene in the progression of radiation-induced diseases. Such intervention may decrease the cancer risk from radiation exposure (Little, 1998). The consequences of the initiation and perpetuation of genomic instability includes such deleterious endpoints as chromosomal rearrangements, delayed mutation, DNA nucleotide repeat instability, cellular transformation, and even cell death.

The third paradigm that radiation-induced changes in gene expression can be demonstrated at very low radiation doses (< 0,5 Gy). This alteration in certain condition can protect cells from effects induced by subsequent higher radiation dose. This phenomena called radiation induced radioadaptive response. Low radiation dose is known may modify the level of damage that induced by subsequent higher radiation dose in human lymphocyte cells (Olivieri, 1984).

This paper deals with radioadaptive response that has been investigated in mice by focusing on the number of total leukocyte counts.

Materials and Methods

Radiation source

Teletherapy is general term used for external exposure with radiation source at a distance from patient. This uses high energy electrons or high-energy X-rays for treatment of deep-seated tumors. High energy γ-emitting radioisotopes such as Cobalt-60 is used for cancer treatment and is the most widely used in teletherapy machines by considering the energy of emitted photons, half life, specific activity, and means of production. Cobalt-60 has a half life of 5.3 years and decays by negative beta emission to metastable ^{60}Ni. This rapidly releases gamma ray of either 1.17 or 1.33 MeV to reach a stable state (Figure 1). According to its energy telecobalt is classified as megavoltage machine because its mean energy is 1.25 MeV (Akhadi, 2000). For irradiating mouse, Co-60 FCC 8000F Teletherapy Machine Sanglah Hospital in Denpasar was used.

Mouse

Eighty mice of 37 days old and 23.79 g of mean body weight were used as control group, mice of 44 days old and 26.66 g of body weight were used for non adaptation group, mice of 45 days old and 26.87 g of mean body weight were for adaptation I group, and mice of 46 days old and 27,18 g of mean body weight were used as Adaptation II group. All mice were controlled and kept in 27^{o}C of room temperature in Vetarinary Center in Denpasar. All experimental processes were done in Technical Service Unit (UPT) Health Laboratory, Ministry of Health, Bali Province, by using hemocytometer and microscope.

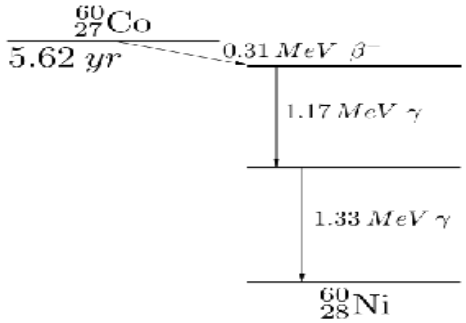

Figure 1. Schema of decay of Co^{60} (K.H. Lieser, 1991).

Mice were group according to Table 1. Mice were maintained in group that each consists of 4 mice in a plastic container with the size of 30x20x10 cm^3 covered with a mesh. Commercial pellet food and drinking water were provided *ad libitum*.

Before collecting the blood, local anesthesia was applied on the tail and a cut was made 1 mm from the tip of the tail using scalpel blade. To examine the number of leukocyte, a drop of blood was taken up to 0.5 mark of the WBC pipette, and then excess blood outside the pipette was wiped with cotton and was proceed according to standard procedure (Depkes, 1991).

Tablel 1. Group of mice with certain number according to the treatment.

No.	Group	No. mice	Irradiation dose (Gy)	Notes
1.	Control	20	No treatment	Age of 0 – 37 days
2.	Non Adaptation	20		Age of 0 – 44 days
	I	4	1.0	
	II	4	1.5	
	III	4	2.0	
	IV	4	2.5	
	V	4	3.0	
3.	Adaptation I	20		Age of 0 – 45 days
	I	4	1.1	Direct irradiation post
	II	4	1.6	adaptation dose of 0.1 Gy
	III	4	2.1	
	IV	4	2.6	
	V	4	3.1	
4.	Adaptation II	20		Age of 0 – 45 days
	I	4	0.1+ 5 h 1.0	Irradiated 5 h post adaptation
	II	4	0.1 + 5 h 1.5	dose of 0.1 Gy
	III	4	0.1 + 5 h 2.0	
	IV	4	0.1 + 5 h 2.5	
	V	4	0.1 + 5 h 3.0	

Blood sampling

Irradiation

Irradiation process was done in 3 steps included non adaptation by expose with doses of 1 Gy, 1.5 Gy, 2 Gy, 2,5 Gy and 3 Gy. And then adaptation I by expose the mice to adaptive dose of 0,1 Gy and directly expose to these doses as non adaptation. Adaptation II was done by exposing the mice to 0,1 Gy and 5 hours later exposed to these doses as non adaptation (Table 1).

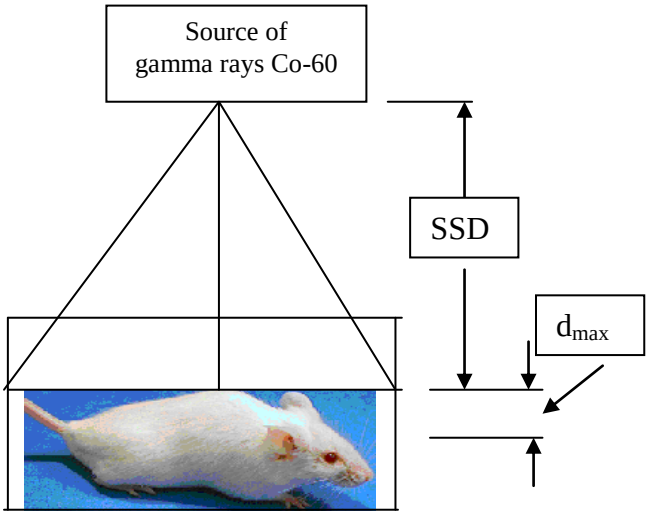

Figure 2. Irradiation condition with constant SSD.

The condition of irradiation was done at SSD (*source to surface distance*) of 80 cm at 2 cm from the upper part of the box (area of 20x20 cm). Dose of irradiation was calculated at the depth (d_{max}) of 0.5 cm (Figure 2).

Results and Discussion

Data of leukocyte number of mice for all treatments are presented in following graph.

It was shown that in general the number of leukocyte was decreased with the increasing of irradiation dose delivered to the mice, except for control which was constant. Compared to non adaptation group, the number of leukocytes in adaptation I was higher and the highest in adaptation II. We can see that adaptation dose of 0.1 Gy caused the alteration in the spontaneous response of leukocyte cells to the next dose exposed, and this alteration was higher when there was an interval time of 5 hours post adaptation dose of 0.1 Gy. It was suspected that there was a recovery taken place in cells exposed to adaptation dose and an adaptation process that related to immune system. Ratio of the leukocyte number of non adaptation, adaptation I and adaptation II compared to control is presented in Table 2 and Figure 4.

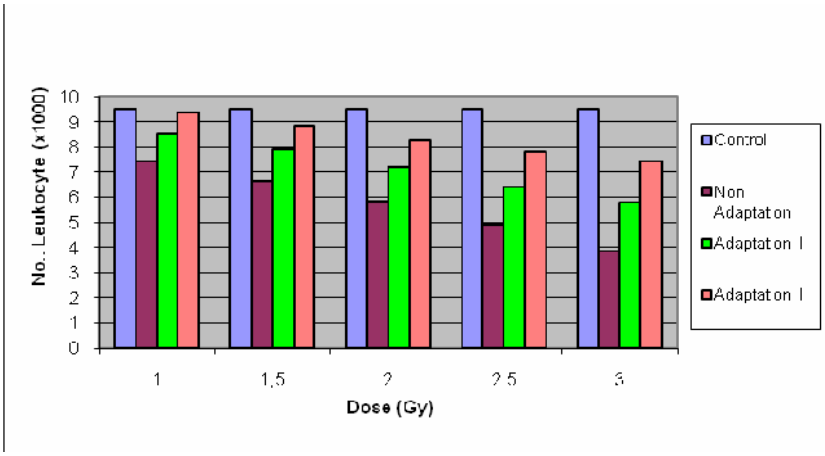

Figure 3. Leukocyte number of mice for all treatments.

Table 2. Ratio of the leukocyte number of non adaptation, adaptation I and adaptation II group compared to control.

Group	Ratio of the leukocyte number to control		
	Non Adaptation	Adaptation I	Adaptation II
I (1 Gy)	0.78	0.90	0.99
II (1.5 Gy)	0.70	0.83	0.93
III (2 Gy)	0.61	0.76	0.87
IV (2.5 Gy)	0.52	0.68	0.82
V (3 Gy)	0.42	0.61	0.78

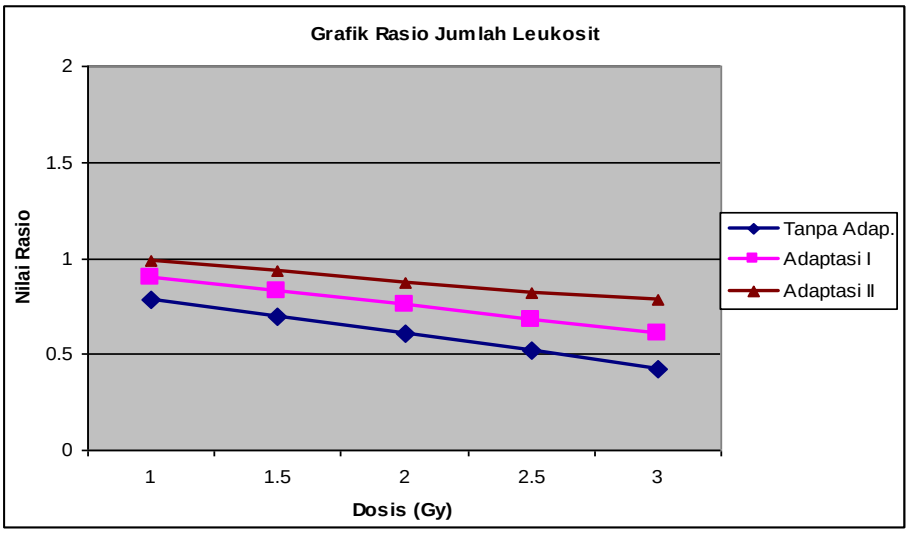

Figure 4. Ratio of the leukocyte number to control.

If X and Y each represents irradiation dose and ratio of the leukocyte number to control, linear equations of all treatment was obtained and presented in Table 3.

Ratio of the decreasing of leukocyte number for non adaptation group was 0.18 to dose of irradiation. The similar tends were also found in adaptation I and adaptation II groups, which were 0.15 and 0.11, respectively, to dose. All linear equation has correlation coefficient (R^2) of 0.99 that means that there is a strong effect of irradiation dose to the number of leukocytes. The decreasing in total leukocyte number is caused by the damage in haemopoietic system (UNEP, 1985). Radiation may induce the inhibition or the arrest in haemopoiesis process that result in the reduction in leukocyte cells in circulation. Doses of 2 – 10 Gy are the dose of haemopoietic syndrome that is signed by reduction of total leukocyte (*leucopenia*) (Hall, 1972).

In this research adaptation dose of 0.1 Gy was effectively increases leukocyte number. Adaptation I and adaptation II gave the increases in leukocyte number by 0.03 and 0.07, respectively. Adaptation

Table 3. Linear equations of leukocyte number to irradiation dose.

No.	Treatment	Linear equation	Correlation (R^2)
1.	Non Adaptation	Y = 0.97 – 0.18 X	0.99
2.	Adaptation I	Y = 1.05 - 0.15 X	0.99
3.	Adaptation II	Y = 1.09 – 0. 11 X	0.99

response is a biological phenomena, the increasing in resistance to radiation may obtained after once or more irradiation with extremely low dose. In radiation field radioresistance after adaptation that alters the next biologic effectiveness with higher dose (Okazaki, 2005; Kadhim, 2004). Low dose was proven increased the capability of human cells, includes *bystander* cells (unexposed cells), to repair chromosomal damage and induces the cells death through apoptosis (Broome, 2002).

In this research challenge doses of 1-3 Gy were delivered to mouse. This dose range is lower than $LD_{50/30}$ for mouse that is 7 Gy. $LD_{50/30}$ (lethal dose) is a dose that caused 50% population of mice death within 30 days (Hall, 1972). Therefore to ensure the long survival of mouse, a maximum dose of 3 Gy was chosen, and the dose rate was 0.996 Gy/minute.

The number of leukocytes of mouse in control group was 9.51 x 10^3/µl that was not different with number of leukocytes found in other research which ranged from 4 to 11 x 10^3/ µl of blood (Effendi, 2003). Irradiation decreased the leukocytes because radiation destroyed parental and precursor cells in bone marrow (bone marrow syndrome), and decreased leukocytes in peripheral blood circulation (Haley, 1965). In normal condition, the loss of blood cell components due to food consumption, infection or aging is balanced by cell production in bone marrow. Radiation may inhibit parental blood cells activity or stop totally their activity which depend on radiation dose. Beside that blood in circulation may also death due to interphase cycle. Radiation may decreased blood cells that also depend on radiosensitivity and life expectation of cells (Hacker, 1984). When nucleic acid (DNA) damage in cells is created as a result of one or more tracks of radiation through a normal cell, the cell will attempt to repair that damage. If the repair is

successful and the DNA restored to its original state, i.e., an error-free repair, then the cell is also restored to normal. In this case, there is no resulting consequence to the cell and hence no resulting risks (Thacker, 1992; Natarajan, 1994).

Data shows that the lowest curve coefficient was seen in non adaptation group (-0.18), and then adaptation I (-0.15) and adaptation II (-0.11). The increasing the curve coefficient the increasing the leukocyte cell response. It gives the opportunity of cell resistance response to next higher dose irradiation (Mitchel, 2000).

The combined exposure (adaptation and challenge dose) resulted in less damaged leukocyte than the single acute 1-3 Gy exposure alone, and when the doses were separated by a 5 h incubation, the resulting leukocyte number was even less. This experiment indicates that the low dose exposure had stimulated the cells to increase their ability to repair the damage, such that the consequences of the second large exposure were reduced. The same respon may found in other organ or tissue. In adaptation II group with adaptation dose of 0.1 Gy and 5 h interval of higher dose, adaptation by the decreasing of leukocyte number had already observed. This finding is in accordance with other research statement that response of leukocyte will be observed within 4 hours post exposure (Hall, 1972).

Results above on leukocyte number in adaptation I and adaptation II groups were also supports the statement in adaptation dose found elsewhere. Dose of adaptation would decreased respon if given before the next higher dose (UNSCEAR, 1994). The range of adaptation dose is 0.05 - 0.2 Gy with interval time of 4 - 18 h before challenge dose (Steffer, 2003), and with low dose of 0.1 Gy will results in apoptosis induction as protective mechanism of cell and loss damaged cells including mutation and efficiency of cell tranformation (Cregan, 1999).

Conclusions

1. In general the number of leukocyte was decreased significantly with the increasing of irradiation dose delivered to the mouse.
2. The adaptive response increased when there was interval time between adaptive and challenge dose. The treatment with adaptation dose of 0.1 Gy before higher challenge doses of 1-3 Gy resulted in the resistance of leukocyte cells.
3. The delay of higher doses for up to 5 h resulted in higher resistance of leukocyte compared to direct exposure of higher doses.
4. Ratio of leukocyte number to control for non adaptation group, adaptation I and adaptation II groups was linearly correlated with irradiation dose, with curve coefficient of -0.18, -0.15, and -0.11, respectively, and with correlation coefficient of R2 = 0.99.

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DISCUSSION

Question:

M. Syaifudin, PTKMR BATAN:

What is the main factor which is affecting/determining the adaptive response of cells to ionizing radiation?.

Answer:

As found in this experiment, the main factor that mostly affecting the cells response is the challenging dose and the interval time between adaptive dose and challenge dose.

In Vivo Evaluation on the Effectiveness of Orally Chloroquine Treatment to the Growth of Gamma Irradiated *Plasmodium berghei* in Mouse

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Abstract. **About 3.3 billion people, half of the world's population, are at risk of malaria. This disease is currently uncontrollable in many areas because of the spread of chloroquine (CQ) resistant parasites and unavailability of effective vaccine. The aim of this research was to determine the effectiveness of CQ in inhibiting the growth of irradiated P. berghei in mouse. Mouse blood containing P. berghei of 2,0 X 10^7 parasites/ml was irradiated with gamma rays at doses of 0, 150 and 200 Gy (doses rate of 380 Gy/hour) and then was injected intraperitoneally to mouse and all mice were given orally 0, 500 and 1000 µg of CQ everyday for 4 days consecutively. The parasitemia (parasite density) in blood and the survival of mouse were observed started on day 3 and repeated every 1-4 days up to 28 days. Results showed that full protection was obtained with 150 and 200 Gy radiation-attenuated parasites in mouse given CQ orally. All these mice were survived during the experiment (more than 30 days post infection). CQ alone did not suppress the growth of parasites as high as that of combination. It was concluded that CQ showed a synergistic effect with irradiation to the parasites growth.**

Keywords : malaria, P. berghei, vaccine, ionizing radiation, chloroquine.

Introduction

Malaria kills more than one million people every year worldwide (Greenwood, 2008). Malaria remains one of the major diseases causing death in Indonesia where about 30,000 people are estimated to die every year (Ministry of Health, 1998). Other report stated that there were 6 million malaria clinical cases and 700 deaths each year in this country (Laihad, 2000). Therefore it urgently needs a malaria vaccine to relieve the human suffering associated with this disease. Many studies demonstrated that a malaria vaccine offering sterile protective immunity was possible. Much of these basic works that were carried out in *P. berghei* rodent model systems has yielded important insights into vaccine–induced protection and has led to the development of a number of candidate vaccines (Nussenzweig et al., 1989; Hoffman et al., 1996a; Hoffman et al. 1996b).

Experimental vaccination with attenuated parasites with ionizing radiation has been shown to offer protection against challenge with malaria parasites (Anonimous, 2006). Nussenzweig et al. demonstrated that irradiated *P. berghei* sporozoites of different ages completely protects against a challenge with normal sporozoites but did not protect against a challenge by erythrocytic stages of the parasite (Nussenzweig et al., 1969). Radiation attenuated sporozoites arrest early in the liver stage development (Scheller et al., 1995), disrupting the normal cycle of the parasite while allowing the host to develop an immune response able to overcome disease upon subsequent challenge.

The long-term fight against malaria involves effective antimalarial therapy for patients such as chloroquine (CQ). CQ was the most important drug for the treatment of malaria for many decades, until widespread resistance led to its replacement, most recently by artemisinin-based combination therapies (Eastman et al., 2009; Wongsrichanalai et al., 2002). CQ's efficacy is thought to lie in its ability to interrupt hematin detoxification in malaria parasites as they grow within their host's red blood cells (Chou et al. 1980; Dorn et al., 1998). Hematin is released in large amounts as the parasite consumes and digests hemoglobin in its digestive food vacuole. Hematin normally is detoxified by polymerization into innocuous crystals of hemozoin pigment and perhaps also by a glutathione-mediated process of destruction (Zhang et al., 1999). CQ binds with hematin in its moxodimer form and also adsorbs to the growing faces of the hemozoin crystals (Dorn et al., 1998; Sullivan et al., 1996; Pagola et al., 2000), disrupting detoxification and poisoning the parasite. CQ affects neither sporozoites nor liver-stages, but kills only asexual forms in erythrocytes once released from the liver into the circulation. Some researchers suggesting that although CQ is not detrimental to the generation of strong antibody responses, it nonetheless exerts a specific effect on cross-presentation rather than a general adjuvant effect (Accapezzato et al., 2005).

However, the lethal form of human malaria that mainly caused by *Plasmodium falciparum* is virtually uncontrollable in many areas because of the development of drug resistance, in particular CQ resistance (Foote et al., 1990). CQ resistance was first reported in Southeast

Asia and South America and has now spread to the vast majority of malaria-endemic countries (Ridley, 2002). Deaths due to malaria are occurring in increasing numbers because of frequent failure of the conventional treatments using CQ against which *P. falciparum* populations have developed a high degree of resistance (Trape, 2001). Therefore it's proposed to combine a vaccine material and therapy with drug as a more reliable treatment option. The present study describes the effectiveness of CQ in suppressing the growth of irradiated *P. berghei* in mice.

Materials and Methods

Parasite and mouse.

Plasmodium berghei (strain Antwerpen-Kasapa, ANKA) infected mouse bloods were received from Eijkman Institute for Molecular Biology, Indonesian Ministry of Research and Technology, Jakarta. Male Swiss-Webster mice (6–8 weeks old) were purchased from National Institute of Health Research and Development, Indonesian Ministry of Health and were housed at the Biomedical Laboratory of The Center for Technology of Radiation Safety and Metrology, National Nuclear Energy Agency of Indonesia (BATAN) animal facility and handled according to institutional guidelines.

Injection of irradiated P. berghei and CQ treatment.

Six to eight week old male mice of Swiss Webster at the start time of experiment were inoculated on day 0 via intraperitoneal injections with 2,0 X 10^7 *P. berghei*/ml of infected blood irradiated with gamma rays at doses of 0, 150 and 200 Gy of Co-60 source (dose rate of 380 Gy/hour). Beginning 2 h after inoculation, once-daily treatments with drug of 500 or 1000 µg (CQ, Sigma-Aldrich) were administered orally for four consecutive days. Drug concentrations were calculated in mg/kg of body weight and adjusted so that aliquots of 0.2 ml would contain the desired dose of CQ. We also included control group mice without drug vehicle alone and irradiation. Starting on day 3 parasitemias in all mice were measured via Giemsa-stained thin smears, and the percent inhibition for each treatment group was calculated for all treatment groups. All data were aggregated from groups of three mice in 2 separate trials.

Results

In vivo interaction of CQ and gamma ray attenuated *P. berghei* had been investigated in this evaluation using mouse as model. The evaluation found that full protection was obtained with 150 and 200 Gy radiation-attenuated parasites in mouse given with 500 and 1000 µg of CQ. All these mice were survived during the experiment (more than 30 days post infection). There was no difference found between 150 and 200 Gy of irradiation and the orally treatment of 500 and 1000 µg of CQ where both were effectively suppressing the growth of parasites. CQ alone did not suppress the growth of parasites as high as that of combination. There was no mouse that had self-cured after the initial infection and were not completely protected, showing no or low immunity in mouse.

For 0 Gy of irradiation, CQ was effectively inhibit the growth of parasites where the prepatent period was 14 days post injection. Contrast to that, there was no inhibition of parasite growing in mouse without CQ treatment (Figure 1). CQ alone did not effectively suppressing the growth of parasites after 15 days of the experiment. A gradual increment in the percentage of parasitemia by over 60 per cent was noted from Day-8 to Day-30 in this control mouse. As found in 0 Gy, CQ was also showed a synergism or effective in suppressing the parasites irradiated with 150 and 200 Gy (Figure 2 and 3). However, it was known that 150 Gy was more effective than 200 Gy due to the shorter survival time of mouse without CQ treatment. This experiment also found that gamma irradiation was known to be much more effective in inhibiting the growth of parasites as shown in Figure 4 where in microscopic observations there were much higher parasitemia in thin blood smear of mouse injected with 0 Gy compared to 200 Gy irradiation even though the mice were treated with 1000 µg CQ.

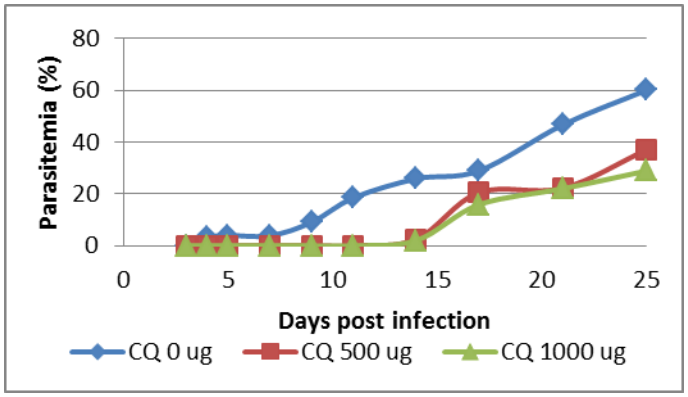

Figure 1. Percentage of parasitemia in mice post infection of 0 Gy irradiated *P. berghei* and Chloroquine (CQ) treatments.

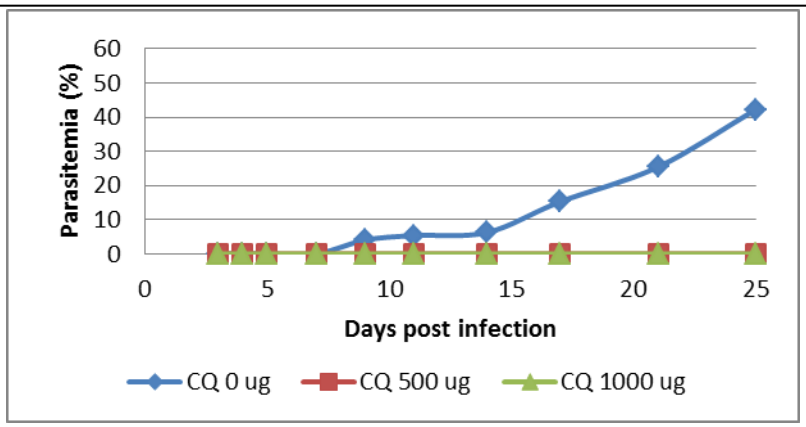

Figure 2. Percentage of parasitemia in mice post infection of 150 Gy irradiated *P. berghei* and Chloroquine (CQ) treatments.

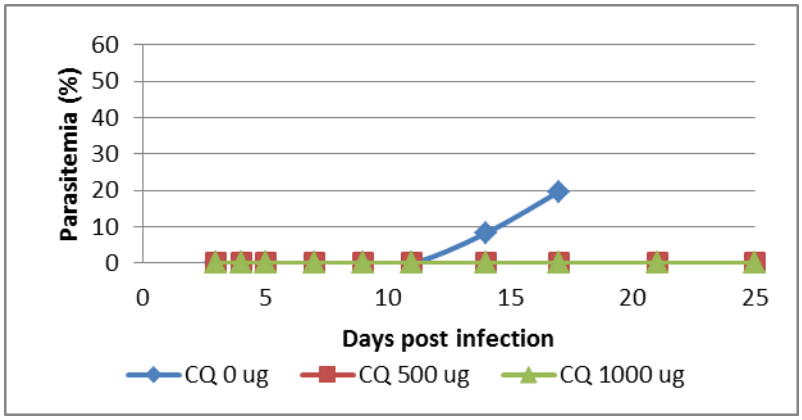

Figure 3. Percentage of parasitemia in mice post infection of 200 Gy irradiated *P. berghei* and Chloroquine (CQ) treatments.

Without CQ treatment Swiss Webster mice infected with non irradiated *P. berghei* developed a patent blood infection 2–3 days post-inoculation and reached almost 60% in the last day of observation. It has been shown that patent blood infection could have a suppressive effect on the immunity directed against parasites. In contrast, however, all mice injected with irradiated *P. berghei* and given orally CQ controlled the infection and eliminated/supressed blood parasites thorough days of experiment. It had been previously established that 10 days of CQ treatment eliminated all erythrocytic parasites in normal mice (Chou et al., 1980). However in this experiment, 4 days treatments were effective in prevent the parasitemia that combined with attenuation by irradiation.

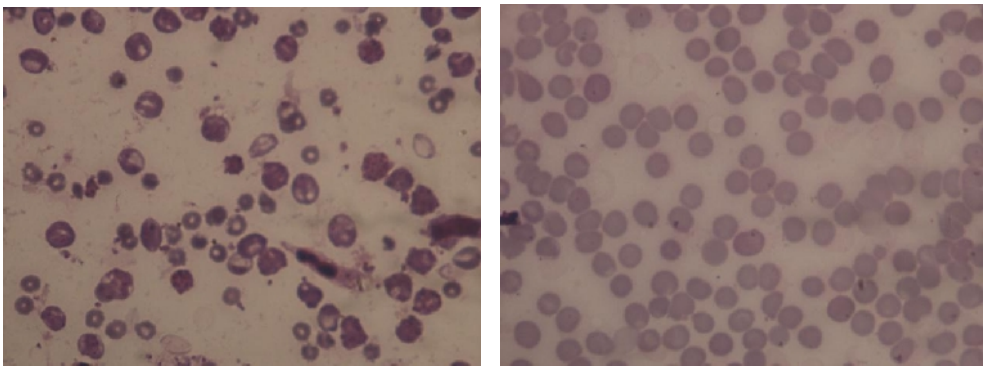

Figure 4. The microscopic observations of thin blood smear of mouse injected with 0 Gy (left) and 200 Gy (right) irradiated *P. berghei* and treated with 1000 µg CQ.

Discussion

Malaria is considered one of the greatest challenges of all the health problems in tropical countries such as Indonesia. While drugs and other interventions are being used to reduce malaria's impact, the disease remains problematic. Although several anti-malarials are available mainly CQ, malaria parasites has gradually developed resistance to nearly all of them, so that anti-malarial drug resistance has emerged as one of the greatest challenges facing malaria control today (Malaria Vaccine Technology Roadmap). An effective and affordable malaria vaccine would seemingly overcome this problem by closing the gap left by other interventions. However, there still no effective malaria vaccine after decades of research and development efforts (Olliaro et al., 1996).

In here we found that CQ did not effectively inhibiting the growth of parasites as gamma rays. This is certainly due to the difference in mechanism of action. Dose of irradiation with its high penetration cause, mainly deoxyribonucleic acid (DNA), the pathogen loses its reproductive ability and virulence, while retaining its viability, metabolic activity and the antigenic profile. Among the metabolic parameters, only protein secretion was changed with the gamma irradiation dose that eliminated cell division (Syaifudin et al., 2011). Therefore, gamma radiation is widely used to inactivate parasite for the preparation of vaccines, appears to create a vaccine that is more effective than so-called "killed" vaccines against disease, and has the added advantage of a longer storage life than "live" vaccines and a strong immune response is induced in the vaccinated host. Its effectiveness was approved by many researchers. A single immunization with 1000 *P. berghei* ANKA strain-irradiated sporozoites can induce protection in some strains of laboratory models such as BALB/c or A/J inbred mice (Jaffe et al., 1980). While CQ drug works by disrupting the mechanism that malaria parasites use to make iron, which is toxic to them, found in the hemoglobin they digest. Normally the parasites avoid digesting the iron by making it insoluble, a process that turns the organs of malaria victims a dark, rusty brown. CQ disrupts this step, causing the parasite to poison itself with iron (Chou et al. 1980; Dorn et al., 1998; Zhang et al., 1999; Sullivan et al., 1996; Foote et al. 1990).

In this experiment, CQ at 500 µg (125 mg/kg body weight) and 1000 µg (250 mg/kg body weight) was proven highly effective in suppressing the growth of parasites. The doses given were higher than that of other experiments conducted by Iwalokun, BA (2008) where 10 mg/kg was used in the study. However, other experiment by Tripathy S et al. (2012) treated the mouse with same doses (250 mg/kg) for 5 days in combination with chitosan-tripolyphosphate nanoparticles to suppress *P. berghei* infection. Experiment by Ridley RG (1997) used similar dose of CQ (285 mg/kg) to study the drug resistance of *P. falciparum* after once treatment. By using CQ-resistant *P. berghei*, Rabinovich SA et al. (1987) studied the suppression of CQ resistance at dose of 250 mg/kg in mouse infected with a microsomal mono-oxygenase inhibitor.

Similar with the present experiment, a more comprehensive evaluation had been conducted by Belnoue E et al. (2008) using irradiated (live) sporozoites of *P. yoelii* under CQ propylaxis and they found that immunity to malaria has long been thought to be stage-specific. In this study they showed that immunization of BALB/c mice with live erythrocytes infected with nonlethal strains of *P. yoelii* under curative CQ cover conferred protection not only against challenge by blood stage parasites but also against sporozoite challenge. They suggested that Ags shared by liver and blood stage parasites can be the foundation for a malaria vaccine that would provide effective protection against both pre-erythrocytic and erythrocytic asexual parasites found in the mammalian host.

Conclusion

In this study we provide evidence through *in vivo* experiments using the suppressive test in *P. berghei*-mouse model that up to 100% suppression of parasitemia following treatment with gamma irradiation and CQ. CQ did not as effective as gamma rays in inhibiting the growth of parasites. This showing that there is the potential value of the combination of irradiation as vaccine development tool and anti-malarial drug for malaria control studies.

Acknowledgements

The technical assistance rendered by staff in the animal care unit of Biomedics Division, Center for Technology of Radiation Safety and Metrology, National Nuclear Energy Agency is sincerely appreciated. This research was conducted under DIPA Grand 2013.

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DISCUSSION

Mrs. Iris, Balitbangkes Jakarta :

Question :

1. Have you got ethical clearance before doing the study?
2. What is the purpose of your study? May be you have a hypothesis, so you want to prove it?

3. The methodology may be not so clear. How you calculate the sample (n) and how you analyze the data to draw the calculation?

Answer :

1. Yes, part of the study was done based on the ethical permission on the use of animal in experimental purposes, and all procedures were done based on the standard procedure with minimal sickness on the animals.
2. The purpose of the study was to get information on the effectiveness of chloroquine (CQ) in suppressing the growth of irradiated *P. berghei* in mice. Our hypothesis was that irradiation and chloroquine has a synergystic action on the suppressing the growth of parasites, and this was proven from this experiment.
3. The methodology was clear and you may read in more detail in the article. We just compare the data of parasitemia from irradiation and CQ treated animal group with positive controls group, both only irradiation and or only CQ.

Humoral Immune Response in Mouse Post Immunization With Irradiated *Plasmodium beghei* and Challenge Test

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Abstract. **Effective immunity to malaria involves humoral and cellular immune components. Antibodies are immune component that plays a role in the humoral immune response. This study aimed to determine the effect of repeated immunization and challenge test for antibody concentration and parasite growth. The study used a complete randomized with 30 Swiss Webster mice that were divided into 3 groups and conducted over 62 days. The radiation doses used were 150 Gy, 175 Gy, and control (0 Gy) and dose of immunization was 1 x 10^7 parasites per ml of *P. berghei*. Immunization replicate was performed 1 and 2 weeks after the first immunization with the same dose of immunization. The challenge test was performed one month after the first immunization by injecting intraperitoneally 1x10^5 infectious parasites/ml into the body of mice. Observation of parasite growth was done every two days and antibody responses observations were made on days 8, 15, 40, 47, and 62. The results obtained as follows, mice were immunized with a dose of 175 Gy had longest prepaten period of 42 days with a mean post-challenge test parasite growth at low and antibody concentrations were observed highest on day 62. There was an enhancing immunity post booster and obtained a protective immunity against infectious parasites. The conclusion that doses of 175 Gy was the optimum dose and immunization debilitating parasite replicates can increase antibody.**

Keywords: Antibodies, P.berghei, immune response, gamma rays, challenge test.

Introduction

Malaria is still a major cause of severe diseases which responsible for million death in tropical countries. In Indonesia, malaria spread almost throughout the island with different degrees of endemicity (Elyazar et al., 2011). This is further complicated by plasmodium resistance to existing drugs, and insecticide-resistant mosquitoes. One alternative to this problem is to bridge precautions against malaria infection by vaccination (Breman et al., 2001).

Attenuation of pathogenic microorganisms is a strategy for vaccine development since it was first discovered by Louis Pasteur (Plotkin et al., 1999). Gamma radiation can be used to inactivate microorganisms for vaccine preparation in addition to the method of heating or chemical inactivation (Eyal and Ferrer, 2006). Gamma radiation for blood-stage parasites aims to weaken the parasite to vaccine preparation. Weak parasites provides an opportunity for the host to develop an immune response that is able to overcome the disease after challenge test.

Parasite that enters the body will soon be facing the body's immune system. An immune response in host against infection malaria is very complex because it involves almost all the components of the immune system both specific and non-specific immunities, humoral and cellular immunities arising in either natural or acquired as a result of immunization. Parasites that enter the blood soon faced by the non-specific immune response and then by the specific immune response (Gilles, 1993). Host immune responses to immunization included response humoral immunity that mediated antibodies and immunity cellular played T lymphocytes. The response of humoral immunity can be known from antibodies secreted (Beeson et al., 2008).

Research in malaria vaccine development that previously have been conducted was to determine an optimal dose of gamma irradiation to weaken the parasites as vaccine ingredients by using a model of *P. berghei* in erythrocytic stage. Effect of irradiation dose was evaluated against the parasitic infection of prepatent period, the percentage of parasitemia, and mortality of mice. Results of previous studies stating that 150-175 Gy is the optimal dose range for γ rays in inhibiting *P.berghei* erithrocytic stage that is characterized by elongation of the prepatent period, low parasitemia and longer life of host (Darlina and Rahardjo, 2012).

Humoral immunity to malaria can be temporary only survive in a short time so that it is necessary to do the second immunization and then followed by evaluation in parasite growth, hematology and serum antibody levels. In this study, evaluation on the long-term immune response and the ability of the parasite to overcome the immune response after challenge test was done.

Materials and Methods

The method used has been conducted based on Ethical Clearance from The Use of Animal Ethics Committee for Health Research, National Institute of Health Research (KEPK-IRB), Ministry of Health of the Republic of Indonesia in 2011.

Animal and vaccines production

Male mice (Swiss Webster), 8-10 weeks old, weighing about 30-35 grams were obtained from the Ministry of Health in Jakarta and quarantined about 7 days in advance before use. Mice were maintained in a fiber glass enclosure with stainless steel lid and given food pellets and drink ad libitum (to taste) (Anonimous, 2011). Mice were divided into 3 treatment groups (mice infected with un irradiated parasite (0 Gy), mice infected with the parasite that irradiated at 150 Gy and mice infected with the parasite that irradiated at 175 Gy). Each treatment group consisted of 20 mice.

Vaccine material was frozen stock *P. berghei* ANKA that was cultured *in vivo* in Swiss mice body in Animal Laboratory of Biomedical Division, PTKMR. Parasite density (parasitemia) in the blood was checked every two days with thin blood smear by cutting mouse tail. When the level of parasitemia had reached above 20%, it was taking blood from the heart after mice were anaesthetized with ether. Blood was collected in mikrosentrifuse tubes and then irradiated at doses of 150 and 175 Gy with a dose rate of 380.5 Gy/h using a Cobalt-60 source at the facility of "IRPASENA Irradiator", Center for Isotope and Radiation Technology Application - BATAN.

Immunization, booster and challenge

Mice were immunized by injecting intraperitoneally 0.3 ml of vaccine material (irradiated with 150 Gy or 175 Gy) that containing ± 1 x 10^7 parasites/ml. Second immunization (*booster*) was conducted one week after the first immunization. Third immunization was performed one week after the second immunization with the same number of irradiated parasites. Challenge test was performed on day 41 by inoculation of $1x10^5$ infectious parasites/ml. Observations of parasitemia and hematology were performed every 2 days until the day 62-nd after infection. The immune response determination was carried out at days 8, 15, 40, 47, and 62 after first immunization. Control mice (0 Gy) was injected intraperitoneally with 0,85 % NaCl.

Counting parasitemia and hematology

Ten uL of peripheral blood was taken from mouse tail end. Two microliters of blood was dissolved in 198 uL solution Hayem for examination the number of erythrocytes. Number of cells were counted using a Neubauer count room under 40X magnification microscope and calculated by the formula:

$$\text{Red blood cells } (mm^3) = \frac{\Sigma \text{ x FP cells}}{0.0025 \text{ mm2 x } 0.1 \text{ mm}}$$

Six microlitter of blood was made for thin smear and was fixed in methanol and stained with 10% Giemsa solution for 30 minutes after that it was washed in running water. Examination and cell counting was done under a microscope at magnification of 1000 X. Parasitemia in blood cells was determined in ± 5000 red blood cells (RBCs) or 10 field of view with the following formula (Ljungstrom et al., 2004).

$$\frac{\text{Number of RBCs infected with the parasite}}{\text{Total number of red blood cells}} \text{ X } 100\%$$

Measurement of antibodies by ELISA

On days 8, 15, 40, 47, and 62 post-immunization, mouse serum was collected by taking blood from the heart function of the groups of mice (0 Gy, 150 Gy, 175 Gy). Blood was centrifuged to separate the serum from the blood cells. Measurement of antibody levels was done by using mouse IgG test kit from DRG and ELISA reader at a wavelength of 450 nm (ELISA Kit, 2012). Scheme of research is shown in Figure 1.

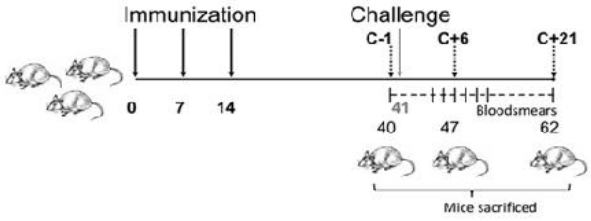

Figure 1. Experimental set-up on mouse post-immunization with irradiated *P. berghei* and challenge test.

Results and Discussion

Parasitemia (parasite density) in blood of control mouse continued to increase until it reaches 49% at day 16 th and all mice died on the 22nd post-infection. Parasitemia in mouse injected with parasites irradiated at doses of 150 Gy was started seen at day 18 post-infection that means that there is a long pre-patent period. Parasitemia in this mouse increased up to 62% at day 38 and all mice died by day 40 post-infection. Parasitemia in blood of mouse injected with irradiated at dose of 175 Gy was started seen at day 44 post-infection or 3 days post-challenge test. Increasing in parasitemia in both treatment groups showed statistically significant differences ($p < 0.05$) (Figure 2). Experiment clearly showed that irradiation was very effective in attenuating parasites and it was more effective after booster. *P. berghei* that was weaken or to be inactive due to radiation was predicted to trigger an immune response in mouse body which can suppress the growth of parasites.

To find out the immune response during infection we then made measurements of the antibody levels during infection in all groups of mice (0, 150 and 175 Gy). Measurement of antibody levels was done by using ELISA. The levels of antibody is expected persist after repeated immunizations (booster) 1 and 2 weeks post first immunization.

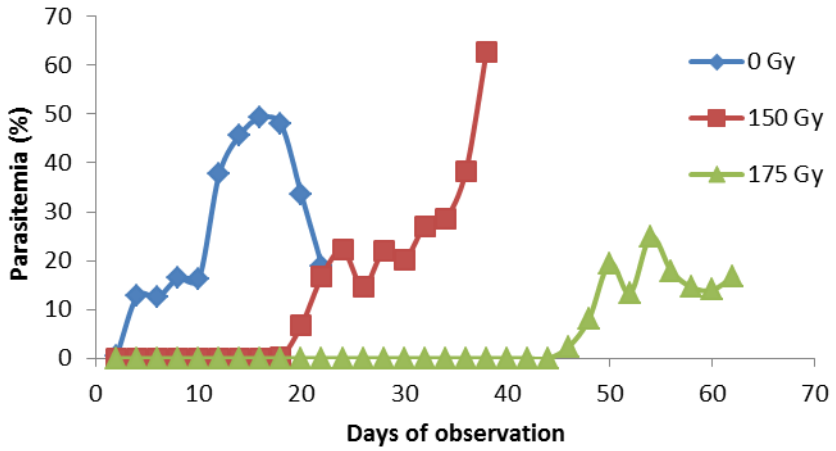

Figure 2. The density of parasitic (parasitemia) after immunization and challenge test.

At day 8th or after the 2nd immunization, the highest antibody levels were in the group of mice *P. berghei* immunized with 150 Gy (Figure 3) and the lowest was found in 175 Gy. The elevated level of antibody in mouse blood serum of 175 Gy was 88,18 ug/ml one day after 3rd immunization (day 15th), whereas in other groups (0 and 150 Gy) the level of antibody was decreased. And then levels of antibody in mouse group 175 Gy was declined one day before challenge (C-1) and six days after challenge (C+6). However at 21 days post challenge (day 62-nd) antibody levels were increased up to 142,6 ug/ml.

The antibodies formed after immunization is expected to inhibit the growth of parasite and reduced symptoms of clinical manifestations of malaria. *P. berghei* in erythrocytic stage multiply in the red blood cells (erythrocytes). An increase in the growth of parasite affecting red blood cells so that the increasing of parasites is generally followed by a decrease in the levels of host erythrocytes (Langhorne et al., 2002). Anemia was a manifestation of a symptom of clinical seen in patients with malaria. To know the influence of repeated immunization against malaria pathology we then determine erythrocyte count after the first immunization until day 8, post 2nd immunization until day 14, and after the 3rd immunization until day 62.

There was an immune response than other antibodies that play a role in suppressing the growth of the parasite. As is known immunity to malaria is complex because it involves almost all the components of the immune system both specific immunity and non-specific, humoral and cellular immunity. Because it is immune to malaria is mediated by a mechanism that acted jointly or separately with protective antibodies. The immunity was apparently played by a mechanism dependent T lymphocyte cells as stated Supargiyono (1995).

The influence of repeated immunization against the number erythrocytes in group of mice immunized with *P.berghei* irradiated with 150 and 175 Gy and control (0 Gy) is presented in Figure 3. The erythrocytes levels of mice injected with infectious parasites (0 Gy) was decreased to 80% after the 15th day (3rd post immunization) and suffering from severe anemia on day 14. Decreased erythrocyte count post 3-rd immunization occurred in 150 Gy group which was about 24% of the first immunization. The erythrocyte counts in the 175 Gy group of mice did not decreased. This proves that repeated immunization with 175 Gy irradiated *P.berghei* could suppress the growth of the parasite and prevent the manifestation of clinical symptoms and mice remained healthy until day 62.

Research on malaria vaccine that used mouse as the model has ever undertaken by other researcher in Indonesia. Wijayanti et al (1997) conducted research on the influence of immunization in mouse with *P. berghei* of erithrocytic stage that had been inactivated chemically. It was known that the partial resistance against parasitic in immunized mice can be seen by the elongation of prepaten period, low parasitemia and the decreasing mortality.

The asexual blood stage of parasites is responsible for the symptoms of the disease. Therefore a significant effort is necessary to develop a vaccine against this stage of the life cycle, which could inhibit parasite growth and consequently prevent or minimize clinical disease (Langhorne et al., 2002). The protection in the previous studies was achieved without adjuvants, an demonstrated the effectiveness of irradiated blood-stage parasites for protection against severe anemia, the most pathogenic consequence of malaria infection. The successful development of an asexual blood stage vaccine is critically dependent upon our understanding of immunity to asexual blood stage parasites (Clyde, 1975).

We also investigated the humoral immune responses induced by immunization with 10^7 non IrrPb or IrrPb blood stage for correlations with the

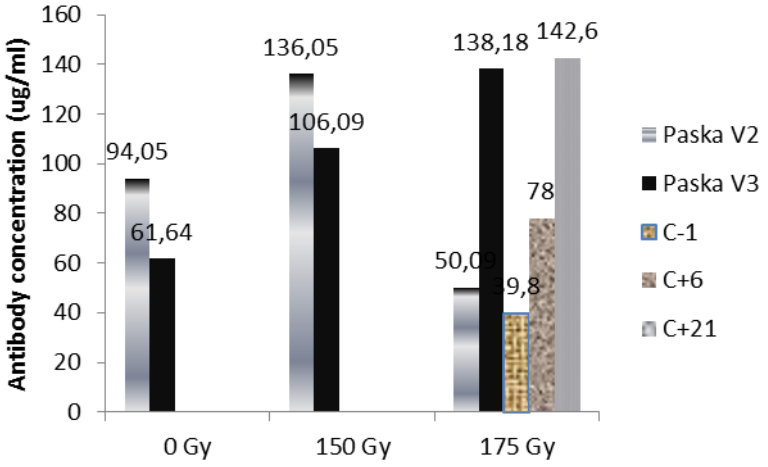

Figure 3. Antibody levels of mice after booster and challenge test.

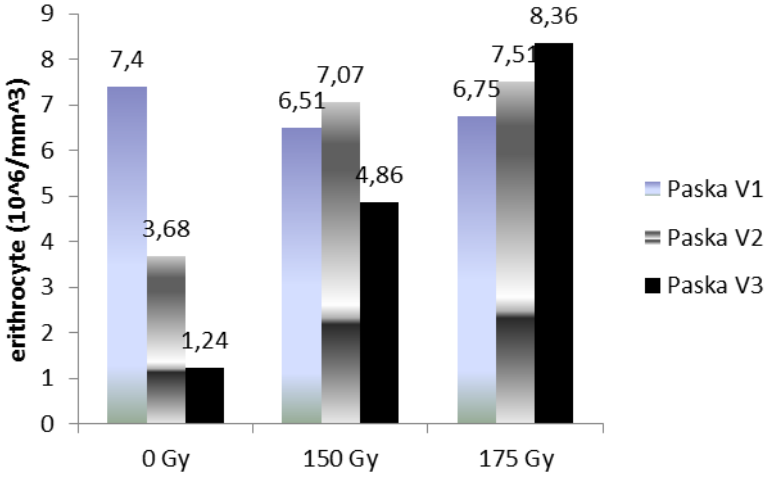

Figure 4. The number of erythrocyte cells of mouse after immunization, booster and challenge test.

protection Swiss mice from parasitemia and severe disease. The antibody responses have been shown to be important to some parasite vaccines and it has been proposed that part of their protective effects may come from an increased clearance of opsonized parasites (Gentoh and Reed, 2007). The attenuated parasites are able to replicate and persist to stimulate the immune system without causing severe disease symptoms. In red cells, parasites appear well located to avoid host responses. Cohen et al (1969) showed that immune response in serum inhibited the development of malaria parasites by acting on free merozoites rather than on the intraerythrocytic form. The responses only at schizont rupture is the parasite directly exposed, for a very brief period, daughter merozoites have to attach to and enter new red cells. Much attention has therefore been given to parasite molecules that interact with the host cells during RBC invasion as potential targets of host immune responses. A number of proteins have been identified on the merozoite surface or in the apical organelles that play a role in RBC invasion and are thought to be targets of immunity. These molecules include MSP, apical merozoite surface antigen 1 (AMA-1), the 175 kDa erythrocyte binding antigen (EBA175) and rhoptry-associated protein 1 (RAP-1), RAP-2 and RESA (Gentoh and Reed, 2007).

The inhibition of erythrocytic invasion by merozoites has been shown to be mediated by spesies specific. IgG and/or IgM antibodies, the effect of anti-merozoites antibodies in serum is inhibitory to erythrocytic invasion. It therefore became apparent that merozoites may be susceptible targets in order to achieve protection of host by either active or passive immunization. The use of such parasites stages in preparation of vaccine is expected to induce protective immunity that is prevents erythrocytic infection (Cohen et al., 1969).

Acquired protective immunity induced by malaria parasites involves both Ab-mediated and cell-mediated immunity. It is well established that B cells and Ab play a crucial role in immunity to malaria. It has been demonstrated that naturally acquired immunity to malaria was seen in individuals living in endemic areas. The degree of protective immunity in humans, monkeys, and mice has been shown to correlate with the level of Ab against asexual blood stage antigen. Acting in collaboration with effector cells such as monocytes and macrophages, they mediate opsonization and Ab-dependent cellular inhibition (Aribot et al., 1996).

Antibody responses directed against surface proteins of the merozoite may function either by blocking RBC invasion or by making the merozoite susceptible to phagocytosis. Parasite antigen-specific Ab play an important role in controlling parasitaemia via Ab-dependent cellular inhibition (ADCI), whereby binding of antibodies to phagocytes leads to inhibition of parasite growth. It has been demonstrated that specific Ab initiate parasite clearance by opsonization, thus enhancing the activity of phagocytic cells or initiating complement-mediated damage (Bouharoun-Tayoun et al., 1990). Since clinical symptoms of malaria manifest only during the blood stage, a vaccine against this stage of the parasite lifecycle would prevent or reduce severity and complications of the disease, and perhaps eliminate malaria if sterile immunity could be achieved.

After initial immunization, a booster injection or booster dose is a re-exposure to the immunizing antigen it is intended to increase immunity against that antigen back to protective levels after it has been shown to have decreased or after a specified period. For these results, in the next time we intended to investigate the antibody responses induced by a multiple immunization with an adjuvant for correlations with irrPb immunized mice with challenge. The protection in the previous studies demonstrate the effectiveness of irradiated blood-stage parasites for protection against severe anemia consequence of malaria infection.

Conclusion

It is concluded that based on humoral immune responses induction, immunization with 10^7 irradiated *P. berghei* blood stage was effectively protect mice from parasitemia and severe anemia. The concentration of antibody was increased in sera from group of mice injected with *P.berghei* exposed 175 Gy irradiations. The level of humoral antibody of immunized mice that measured by ELISA was higher than that of the non-immunized mice, where the highest level was observed in mouse immunized with 175 Gy (142,6 µg/ml) at day 62.

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DISCUSSION

Question :

Prof. Ohtsura Niwa, Fukushima Medical University, Japan :

How is it possible that for immunization you inject the mouse with irradiated parasites at number of, for example, 700 of attenuated (non active) parasites and 300 parasites death due to irradiation, and in the another hand it is compared if you inject mouse with only 700 attenuated parasites?

Answer :

Yes it is possible to do that but it is so difficult to determine or calculate the exact number of 700 of attenuated (non active) parasites and 300 parasites death due to irradiation. In this research and also in many other experiments the parasites at certain same number of parasites are irradiated with 0 and 150 Gy of gamma rays to attenuate them and it was proven efective to induce immune respon.

The Influence of Booster and Challenge with Gamma Irradiated *Plasmodium berghei* of Erythrocytic Stage on the Erythrocyte and Leukocyte Counts in Mouse Blood

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Abstract. **Malaria is one of the major human infectious diseases in many endemic countries including Indonesia. Many clinical symptoms of malaria are associated with alterations in certain haematological parameters/counts. This study aimed to determine the effect of booster (repeated immunization) and challenge test on the erythrocyte and leukocyte counts as well as parasite growth in mouse blood. Thirty Swiss Webster mice were divided into 3 groups each were injected intraperitoneally with 0, 150, 175 Gy irradiated 1 x 10^7 parasites per ml. Booster (second immunization) was done 1 and 2 weeks after the first immunization with the same number of parasites. The challenge test was performed one month after the first immunization by injecting intraperitoneally 1x10^5/ml infectious parasites of *P. berghei* into the body of mice. Observation of erythrocyte and leukocyte counts and parasite growth was done every two days up to 62 days post first injection. The results showed that mice immunized with irradiated parasites have longer prepaten period of parasitemia and steady hematology concentrations, mainly on higher dose of irradiation used. They were also survived for longer time. The conclusion that can be drawn from this research is that dose of 175 Gy is the optimum dose of irradiation based on the hematological counts that related to the increased antibody.**

Keywords: malaria vaccine, gamma rays, P.berghei, booster, challenge test, haematologic counts

Introduction

Malaria is one of the major human infectious diseases in over 100 endemic countries, there being approximately 300 million clinical cases and 2 million fatalities per year (World Health Organization, 2008). Prompt and accurate diagnosis is one of the keys for effective disease management, being one of the main interventions of the global malaria control strategy (World Health Organization, 2006). For prevention of the spread of malaria, it is necessary to develop an early, sensitive, accurate and conventional diagnosis system (Yatsushiro et al., 2010). Attempts to control the spread of the disease have been severely impeded by the emergence of drug-resistant parasites as well as insecticide-resistant mosquitoes and by inadequate health-case infrastructures in those countries that are the hardest hit by the disease. Among the potential control measures that have been given high priority by national and international health organizations, the development of a vaccine against malaria is recognized as one of the most promising and cost-effective addition to the arsenal of current malaria-control measures (Cohen, 2012). To support this, a safe and effective vaccine is urgently required to enhance existing malaria control measures.

Gamma radiation is widely used by many researchers to inactivate parasite for the preparation of vaccines, instead of traditional heat or chemical methods of inactivation, appears to create a vaccine that is more effective than so-called "killed" vaccines against disease, and has the added advantage of a longer storage life than "live" vaccines. Irradiation is a technically simple process that retains structural features of the microbial pathogen without destroying the natural antigens or the intrinsic adjuvants. Therefore, a strong immune response is induced in the vaccinated host. Irradiation destroys the nucleic acid, making the microorganism unable to replicate so it cannot establish an infection, but some residual metabolic activity may survive, so the irradiated microorganism can still find its natural target in the host (Anonymous, 2006).

Many clinical symptoms of malaria are associated with alterations in certain haematological parameters during acute and subclinical infections (Orago et al., 2001). Partial immunity, whether induced by repeated immunization with whole parasites or with vaccine-seems important to the development of anemia. Blood count abnormalities are a recognized feature of many parasitic infections and immunizations but little is known about the haematological effects of malaria vaccination (Cummins et al., 1998). This led to a series of studies

in the diagnosis of malaria especially in developing its vaccine.

Blood is the most easily accessible diagnostic tissue of the body. Changes in haematological parameters are likely to be influenced by any disease condition such as malaria which affects the haemopoetic physiology at any level. This is likely to happen with an endemic disease such as malaria that affects the host homeostasis at various fronts resulting in a myriad of clinical presentation. Haematological changes in malaria, such as anaemia, thrombocytopaenia and leucocytosis or leucopaenia and erythropaenia are well recognized. The extent of these alterations varies with level of malaria endemicity, background haemoglobinopathy, nutritional status, demographic factors, and malaria immunity (Erhart et al., 2004; Price et al., 2001; Wickramasinghe and Abdalla, 2000).

Materials and Methods

The method used in this research has been obtained with the ethical clearance from The Use of Animal Ethics Committee for Health Research of National Institute of Health Research and Development (Litbangkes), Ministry of Health of the Republic of Indonesia in 2011.

Animal materials and vaccines production

Mice (Swiss Webster) male, 8-10 weeks old, weighing about 30-35 grams obtained from the Ministry of Health in Jakarta and quarantined for about 7 days in advance before use. Mice were maintained in a fiber glass cage with stainless steel lid and given food pellets and drink ad libitum as guided in standard protocol. Mice were divided into 3 treatment groups (mice infected with the parasite was not irradiated (0 Gy), mice infected with the parasite that irradiated with 150 Gy and mice infected with the parasite that irradiated with 175 Gy. Each treatment group consists of 20 mice.

Vaccine material is frozen stock *Plasmodium berghei* ANKA that cultured *in vivo* in Swiss mice body maintained in Biomedical Animal Laboratory of the Center. Parasite density (parasitemia) in the blood was checked every two days with cutting edge of mouse tail. When the level of parasitemia of mice has reached above 20%, the mouse were sacrified and blood from the heart was taken. Blood was collected in microsentrifuge tubes and then was irradiated at doses of 0 (control), 150 and 175 Gy with a dose rate of 380.5 Gy/h using a Cobalt-60 source at the facility "IRPASENA Irradiator", Center for Isotope and Radiation Technology Application - BATAN.

Immunization, Booster and challenge

Three groups of mice were injected intraperitoneally (IP) with 0.3 ml of vaccine material of these irradiated *P. bergehi* at ± 1 x 10^7 parasites. Second injection (*booster*) was conducted one week after the first injection with the same irradiated parasites number. Third injection was performed one week after the first booster with atthe same number of parasites. Challenge test was performed on day 41 by IP inoculating 1x 10^5 infectious parasites. Observations of parasitemia and hematology ewere performed every 2 days until the day 62 and after first injection of parasites.

Counting parasitemia and hematology

Ten microlitter of peripheral blood mouse was taken from the tail end. Two microliters of blood was dissolved in 198 uL Hayem solution for examination of the number of erythrocytes. Number of cells were counted using a Neubauer count room under 40X magnification microscope as standard protocol and calculated by the formula :

Red blood cells (mm^3)= (Σ x FP cells)/(0.0025 mm^2 x 0.1 mm)

Six microlitter of blood was made a thin smear on glass plate and fixed with methanol and then stained with 10% Giemsa solution for 30 minutes followed by washing in running water. Examination of cell counts was done under a microscope at magnification of 100 X. The density of parasitemia in blood cells of mouse infected with the parasite in ± 5000 red blood cells or 10 fields of view is calculated with the formula published elsewhere (Hendrix, 2012).

Results

Results of experiment are presented in Figures 1, 2 and 3. In Figure 1 it can be shown that the number of erythrocytes of mouse injected with irradiated parasites is higher than that of control (0 Gy) except for the first 2 days of observation. The erythrocyte counts of mouse group injected with 175 Gy irradiated parasites were thoroughly higher than other groups during the experimental period. Their counts were also within normal range up to 62 days of observations. The survival time of this group is also much longer than others. The life time of 175 Gy irradiated *P. berghei* injected mouse was almost 2 and 3 times longer compared to control and 150 Gy group, respectively.

In contrast to the above finding, the number of leukocytes of control group was higher than that of 150 and 175 Gy injected parasites except for days 10-18.

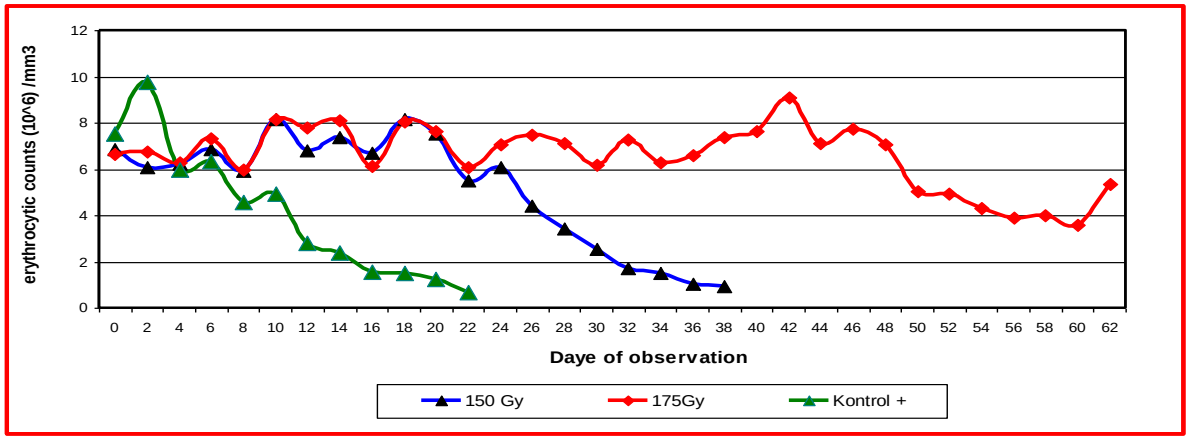

Figure 1. The mean erythrocytic counts (/mm3) in mouse blood at days post booster and challenge test with 150 and 175 Gy irradiated *P. berghei* of ANKA strain.

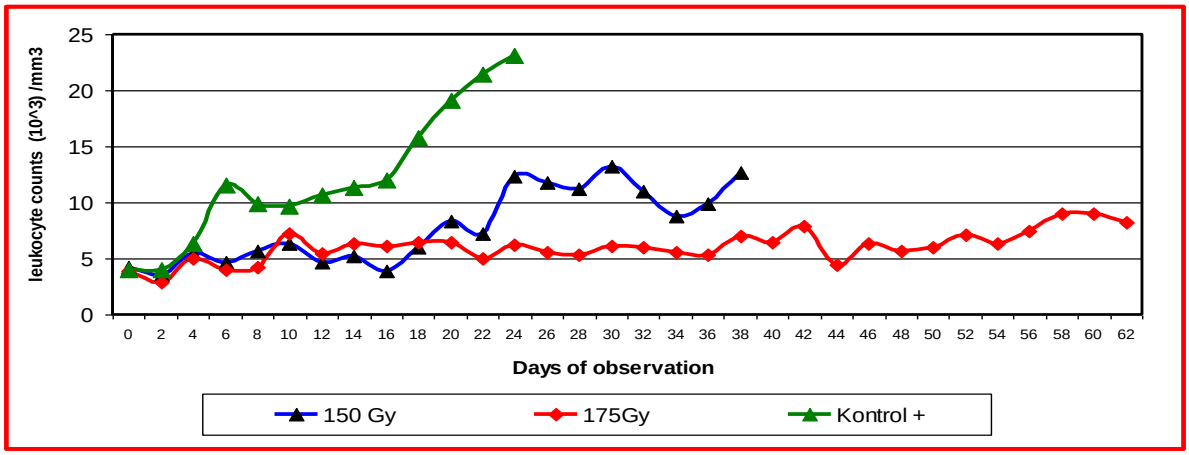

Figure 2. The mean leukocyte counts (/mm3) in mouse blood at days post booster and challenge test with 150 and 175 Gy irradiated *P. berghei* of ANKA strain.

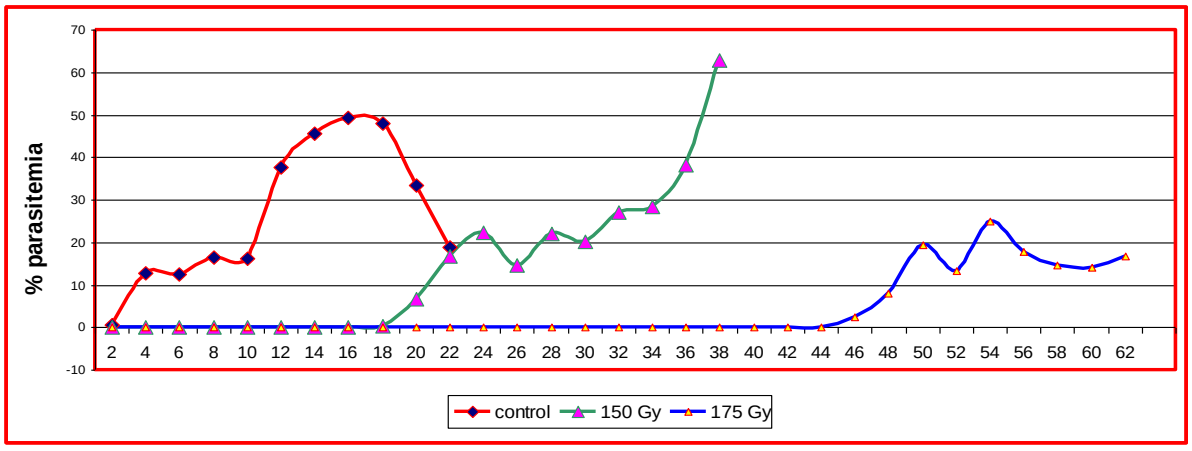

Figure 3. The parasite density (parasitemia) in mouse blood at days post booster and challenge test with 150 and 175 Gy irradiated *P. berghei* of ANKA strain.

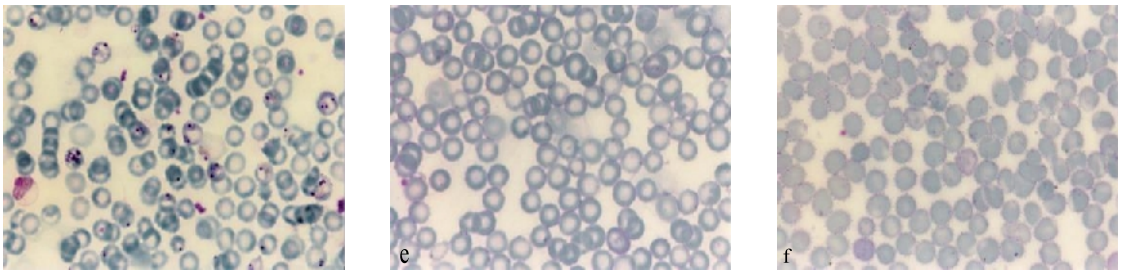

Figure 4. The parasite density (parasitemia) in mouse blood at days post booster and challenge test with 150 and 175 Gy irradiated *P. berghei* of ANKA strain.

The same result was also found in parasitemia where the parasite density in mouse blood injected with irradiated *P. berghei* followed by booster was low or even zero and its prepatent period was about 42 days, compared to 18 days for150 Gy and 2 days for control mice (Figure 3). Microscopic views of thin blood smear of mouse at days post booster and challenge test with 0, 150 and 175 Gy irradiated *P. berghei* of ANKA strain are presented in Figure 4.

Discussion

In this simple experiment, only 2 types of haematology parameters (erythrocyte and leukocyte) were observed in blood of mouse after booster and challenge test. These two cells have an important function in the body. Erythrocytes are red blood cells and contain hemoglobin and their main function is to carry oxygen to all the body cells. Leukocytes are white blood cells of which there are many types includes B lymphocytes, T cell lymphocytes, natural killer cells, etc. The shape of erythrocytes is ideal for this function. Erythrocytes, which represent the most numerous cell type in the body die at a rapid rate, 2-3 million erythrocytes die every second. Erythrocyte production must equal erythrocyte death or the cell population would decline (Anthea et al., 1993). Leukocytes are white blood cells that are part of the body's immune system. They destroy and remove old or aberrant cells and cellular debris, as well as attack infectious agents (pathogens) and foreign substances. Leukocytes is the collective word for all these and have specific functions. This also functioning in the protection from pathogens, bacteria, viruses as the killing of cancerous cells (Hajdu, 2003).

One research on malaria patients was done on the observation of many haematologic parameters such as platelets, lymphocytes, eosinophils, red blood cell count and haemoglobin (Hb), absolute monocyte and neutrophil counts, and platelet volume (MPV) that were higher/lower in malaria infected children compared to non-malaria infected children (Maina et al., 2010). Children infected with *P. falciparum* malaria exhibited important changes in some haematological parameters with low platelet count and haemoglobin concentration being the two most important predictors of malaria infection in children in our study area. When used in combination with other clinical and microscopy, these parameters could improve malaria diagnosis in sub-patent cases.

Another study confirms that haematological abnormalities/alterations considered to be a hallmark in malaria infection diagnoses and its vaccine development and more pronounced in *P. falciparum* infection. The abnormalities or alterations previously reported include changes in haemoglobin, leucocyte count, platelet abnormalities resulting in defective thromboplastin, and disseminated intravascular coagulation in malaria patients that was done by several researches (Reyburn et al., 2007; Wickramasinghe and Abdalla, 2000). Anaemia is one of the most common complications in malaria.

As done in this experiment, prime-boost immunizations/injection with irradiated parasites can induce strong hematologic responses and longer prepatent period as well the survival life of mouse. The precise pattern and levels of cellular immunity required for protection against malaria remain unknown. A major obstacle in malaria vaccine research is the identification of correlates of response to vaccination and protection against malaria (Bruna-Romero et al., 2001).

Conclusions

The booster was effectively suppresses the parasites growth and dose of 175 Gy is the optimum dose of irradiation based on the hematological counts

that related to the increased antibody. They were also survived for a much longer time than control.

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DISCUSSION

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Question:
What is the reason of your experiment with booster and challenge with vaccine materials on mouse?

Answer:
The most effective vaccination protocol against malaria in animals and also in humans consists of the booster (repeated immunization) with radiation-attenuated parasites. This immunization not only induces neutralizing antibodies, but also elicits effective cell-mediated immunity. Challenge is also important to test the effectiveness of immunization and booster treatment.

The Assessment of Source Term, Consequences and Risks on Fukushima Accident

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Abstract: **The core meltdown in Fukushima Daiichi nuclear power plants due to loss of electric power raises the consequences of radiation accidents to the environment. The nuclear accident at the Fukushima Daiichi nuclear power plant has compelled the international community to consider whether any attempt is being done to ensure nuclear safety. This accident was classified into severe accident. The objective of this assesment is to estimate the extent and severity of accidents for particular source term, to determine the effects and consequences of accident, and the risk accepted by society and the environment. This assesment includes source and magnitude determination of the fission products release from the reactor. The assessment is started with the calculation of source term based on BWR core inventory for 3 units reactor, core damage assumptions, and the radioactive release model from the core to the environment. Inventory calculations were performed using the ORIGEN2 computer code. The consequences, impacts, counter measure, and the risk accepted by public and the environment were calculated using the EC-MARIA Cosyma computer code based on source term and meteorological conditions at the time of the accident in the site. The consequences of source terms, and risks of this accident was influenced by postulations and assumptions made to the accident. Although the meteorological data at the Fukushima site are limited, the calculation results showed that no significant difference compared to the results from other researchers. The results of this assessment indicated that the assumptions to simulate the severe accidents are more pessimistic than other assessment.**

Keywords: Source term, radiological consequences, risk, Fukushima accident.

Introduction

The total core meltdown in Fukushima Daiichi NPP has placed international energy policy at a crossroads. The global renaissance of nuclear energy hailed for decades has failed to materialize and following the nuclear disaster in Japan. It has become even more unlikely that nuclear energy will play an important role in the global energy mix over the long term. The accident at the Fukushima Daiichi nuclear power plant has compelled the international community to consider whether any attempt is being done to ensure nuclear safety. The Fukushima accidents was classified into severe accident. The air is a major exposure pathway by which radioactive materials discharged under normal operation or accident could be dispersed in the air and transported to locations where they may reach the public (SG, 2002).

Severe accidents and accident management in the design of nuclear power plants (S.G, 2009): "Certain very low probability plant states that are beyond design basis accident conditions and which may arise owing to multiple failures of safety systems leading to significant core degradation may jeopardize the integrity of many or all of the barriers to the release of radioactive material". The Fukushima accident has caused the impact, consequences, risks and health impacts to people who live in the vicinity of the site and off-site. The assessment of an accident impact is necessary to know the severity of the impact. Impact assessment in this study can be used as a comparison with other assessment regarding to Fukushima accident.

The objective of this assessment is to estimate the extent and severity of accidents in particular source term, to determine the effects of accident, the consequence, and the risk accepted by society and the environment. This assessment includes the source and magnitude of the fission products release from the reactor. The assessment started with the calculation of source term base on BWR core inventory for 3 Units reactor, core damage assumptions, and the radioactive releases model from the core to the environment. Inventory calculations were performed using the ORIGEN2 computer code. The consequences, impacts, counter measures, and the risks accepted by public and the environment were calculated using the EC-MARIA Cosyma computer code based on source term and meteorological conditions at the time of the Fukushima site.

Calculation Methodology

The assessment consists of several calculations, that are core inventory calculation (3 units reactor in Fukushima Daiichi), source term calculation, consequences calculation (individual dose and collective dose), countermeasures, and risks.

Table 1. Fukushima Dai-ichi Nuclear Power Station.

Unit	Electric power output (MWe) / Thermal Power output (MW)	Type of Reactor	Number of fuel rods	
			Core	Spent fuel
Unit-1	460 / 1380	BWR-3	400	292
Unit-2	784 / 2381	BWR-4	548	587
Unit-3	784 / 2381	BWR-4	548	514
Unit-4	784 / 2381	BWR-4	0	1331

Core Inventory Calculation

The inventory core calculation was performed using the Origen 2 computer code (2002) based on JAIF data as shown in Table 1 (JAIF, 2011). The amount of uranium and average burn-up in every reactor are used as input data. Calculations in the Origen computer code were performed according to equation 1).

The fraction of fission products leaking out for time *t*, is given by (Willer, 2005) :

$$\frac{L(t)}{B_0} = \frac{X}{2400} \cdot \frac{1}{(\lambda + \frac{X}{2400})} \cdot \left[1 - \exp\left[-(\lambda + \frac{X}{2400})t\right]\right] \quad \text{... 1)}$$

Where,

B_0 (atoms) : Amount of radioactive material initially dispersed throughout the volume.

X (%/day) : Leakage rate (% per day) from the volume.

λ ($hour^{-1}$) : Decay constant for the radioactive material.

This could be caused by radioactive decay or deposition/plate-out on the surfaces of the volume.

L(t) (atoms) : Amount of radioactive material released from the volume as a function of time, *t*.

t (hour) : Time

Fission Product Release, consequences, risk and countermeasure Calculation

Fission product release or source term is a radioactive material released into the environment from the reactor accident, so it will have an impact to the environment and public health. Figure 1 shows the assessment model of fission product release, used in this study. First, fission products release to the air were calculated using the reactor core failure data in Table 2. By assuming a severe accident according to IRSN (2007) and NUREG (1995), then the source term calculations were carried out using data in Table 1 and Table 2.

The calculation based on local meteorological as input data were carried out for air dispersion. The calculations of fission products release in the air, consequence, risk, and countermeasures at NPP site were carried out using PC-Cosyma computer code (1995).

Using local meteorological data needed in the calculation of air dispersion, then the calculation were carried out to estimate the amount of fission products released in the air and deposited on the ground surface, within various pathways (cloudshine, groundshine, and via food chain). Then, the consequences, risks, and countermeasures to environmental site were calculated.

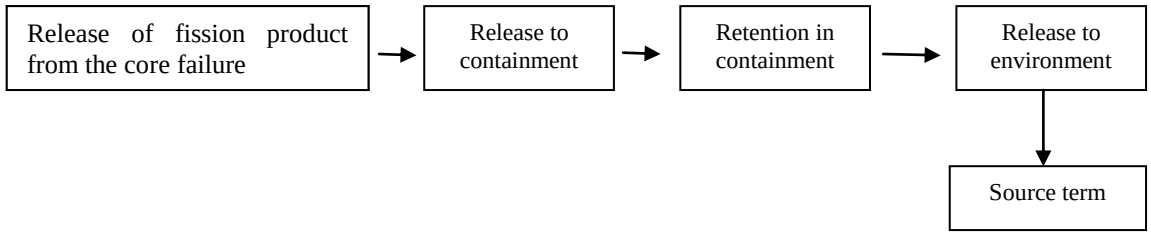

Figure 1. The assessment model for fission product release.

Tabel 2. Information on Status of Nuclear Power Plants in Fukushima.

Status	**Fukushima Dai-ichi Nuclear Power Station**		
Unit	1	2	3
Electric Power (MWe) / Thermal Power output (MW) *)	460 / 1380	784 / 2381	784 / 2381
Fuel assemblies loaded in Core*)	400	548	548
Core and Fuel Integrity (Loaded fuel assemblies) *)	Damaged 70%	Damaged 30%	Damaged 25%
Containment Vessel structural integrity*)	Not Damaged	Damage and Leakage	Not damaged
The released of fission product from the failed core*)	70 %	30 %	25 %
The released to containment **)	50 % I, other 1 %	50 % I, other 1 %	50 % I, other 1 %
The retention in containment **)	No spray	No spray	No spray
Release to environment **)	With filter HEPA	Without filter HEPA	With filter HEPA

*) JAIF

**) IRSN (2007), NUREG (1995)

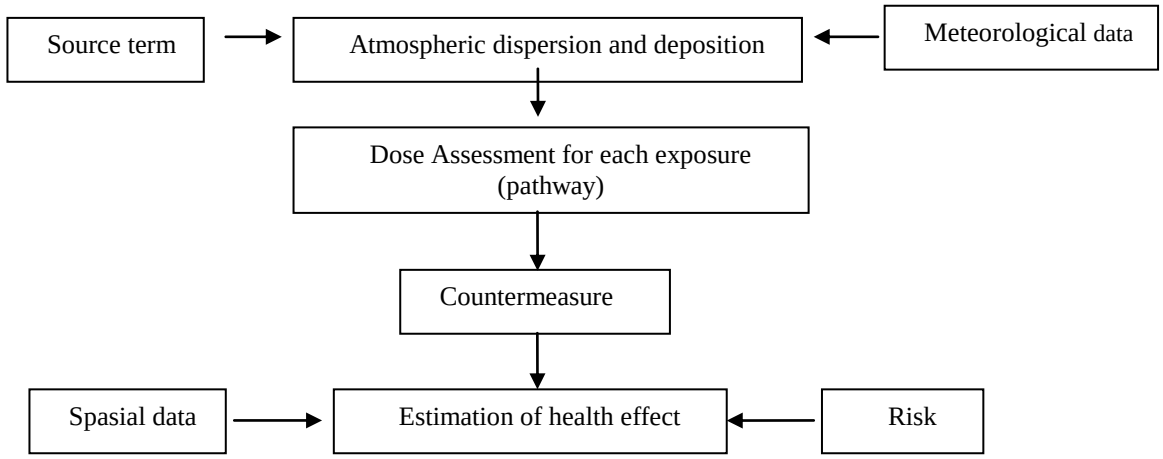

Figure 2. The assessment modelling for consequences and risks.

Figure 2 shows flow chart of health effect assessment. PC-COSYMA is a dose consequence assessment computer code according to segmented Gaussian diffusion model using source data derived from accident scenarios. This computer code can be used to calculate :

- hourly changes in meteorological conditions,
- horizontal and vertical dispersion parameters which vary with the surface roughness as a function of release height,
- plume rise and building induced downwash.
- radioactive decay and daughter in-growth.

This model is similar to the simple puff model but it allows changing certain parameters both in space and time. In the segmented plume approach, the plume is divided into independent elements (plume segments or sections) with has initial conditions. In this model, the plume element depends on emission and meteorological condition which vary in time (function of time) along their motion. Segments, in fact, are sections of the Gaussian plume and each segment generates a concentration field that is still basically calculated by the Gaussian equation described in previous section, and it represents the contribution of the entire virtual plume passing through that segment. The puff segments can follow 3-dimensional wind trajectories and could be used for distances up to 1000 km (Clark, 2005).

Results and Discussions

Core Inventory

Table 3 shows the calculation results of core inventory. Core inventory activity on the Unit-4 was not estimated because at the time of the accident there was no fuel. Eight major groups of fission products should be considered in the power reactors are; Group 1: Xe, Kr; Group 2: I, Br ; Group 3: Cs, Rb; Group 4: Te, Sb, Se; Group 5: Ba, Sr; Group 6: Ru, Rh, Pd, Mo, Tc; Group 7: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am; and Group 8: Ce, Pu, Np.

Table 3 shows the core inventory activity of 3 Unit reactors. It shows that Unit 2 and 3 have a higher activity than Unit 1. As shown in Table 1, reactor core in unit 2 and 3 have 548 fuel assemblies in each core, while unit 1 has 400 fuel assemblies. Fission product activities in the core are influenced by the weight of uranium, that is the higher amount of uranium, the higher the activity. Thus, the same amount of uranium contained in the fuel assemblies in Unit 2 and Unit 3 which greater than Unit 1, therefor the activity of fission products in the reactor core Units 2 and Unit 3 are higher than Unit 1. Moreover, Fission product activity in the core inventory also depends on reactor power or burn-ups, and irradiation time.

Table 3. Fission product activity in core inventory.

Nuclide	Core Inventory Activity (Bq)		
	Unit-1	Unit-2	Unit-3
Kr-85	1.91E+16	3.34E+16	3.34E+16
Kr-85m	2.99E+17	5.18 E+17	5.18E+17
Kr-88	7.94E+17	1.37E+18	1.37E+18
Xe-133	2.48E+18	4.29E+18	4.29E+18
Xe-135	7.24E+17	1.25E+18	1.25E+18
I-131	1.27E+18	2.19E+18	2.19E+18
I-132	1.82E+18	3.15E+18	3.15E+18
I-133	2.55E+18	4.41E+18	4.41E+18
I-135	2.38E+18	4.12E+18	4.12E+18
Te-132	1.78E+18	3.07E+18	3.07E+18
Cs-134	3.08E+17	5.32E+17	5.32E+17
Cs-137	2.06E+17	3.56E+17	3.56E+17
Sr-90	1.36E+17	2.35E+17	2.35E+17
Ru-106	9.11E+17	1.57E+18	1.57E+18
Ba-140	2.17E+18	3.75E+18	3.75E+18
Ce-144	1.64E+18	2.84E+18	2.84E+18

Fission Product Release (source term)

Table 4 shows the calculation results of fission products release under severe accident scenarios, using data described on Table 1 and Table 2. It shows that the highest activity is the source term of Unit 1. Comparing to the activity inventory in Table 3, the source term in unit 3 should be the smallest one. In the source term calculation, the results depend on the core integrity and

core failure. Because the unit 1 reactor is the most severe in condition (core failure 70%), thus, even unit 1 is the smallest inventory activity, it has the largest fission products leaked into the air. While, though Unit 2 and Unit 3 reactors have the same activity on the core inventory, but they have source term with different activities. It caused by the difference of core failure. The core failure in unit 2 and unit 3 are 30% and 20%, respectively. In addition, the containment integrity (Table 2) also influences the source term release. Containment integrity corresponds to the function of the chimney HEPA filter. If the containment is damaged, it can be considered that the filter is not functioning properly. Though, in normal condition, HEPA filter could capture 90% iodine and 99% for the other fission product.

Table 4 shows that the activity of total source term for I-131 is 3.21E+17 Bq. While, NESA (2011) result is 1.60E+17 Bq, or 1.50E+17 Bq (NSC, 2011). Comparing to these three results, the calculation performed in this paper gave a higher source term, although it still within the same order number. This may imply that the assumption of the calculations in this paper is more pessimistic than other calculations. But, regarding to Cs-137 nuclides (the activity is 1.97E+15 Bq) there is a significant differences compared to 1.50E+16 Bq. (NESA, 2011) or 1.10E+17 Bq (NSC, 2011). In this calculation, besides iodine nuclide group, the safety features function are still assumed running properly, so that only 1% of the fission products of Cs-137 released to containment.

The differences arise in these results (current study, NESA and NSC), are caused by the postulations, assumptions, and calculation models which had being taken. Source term calculations depend on postulations, assumptions and calculation models. The level of confidence in the condition that occurs will also result in different calculations such as pessimistic calculations or optimistic calculations.

Dose calculation

Based on these radiological observations, the calculation of fission products release in the air, consequences, countermeasures, and risks estimation at Fukushima Daiichi area were performed. Table 5 shows dose calculation results which were performed by using meteorological data at Fukushima Daiichi area when the doses measured, source term data in Table 4, and assessment model in Figure 2.

The dose calculations were performed for short-term individual effective dose and long-term individual effective dose for each radius. Collective dose was determined by assuming a single homogeneous population. Exposures in long-term individual doses were modeled through a cloudshine pathway and groundshine. While the calculations for the long-term doses were modeled through groundshine pathway (via the food chain). The potential exposure situation for a nuclear reactor facility was not a typical accident affected by the fuel in the core which releases radioactive material to the environment. Thus the effects to the public and types of consequences may be different.

According to The Basic policy in the ICRP (2007) recommendation, in the emergency activities such as nuclear accident the annual dose to the people is allowed up to 100 mSv. It's defined as emergency exposure situation. Post emergency situation, the annual dose is limited within 20 mSv for the existing exposure situation for long-term exposure. And finally, additional exposure dose reduce less than 1 mSv/y should be achieved for longterm goal (Yoshida, 2013). The dose data in Table 5, indicates a short dose in the range of 24-48 hours after the accident and calculated for each unit. Dose in the exclusion area (within 800 m) for Unit 1 is 3.46E-01 Sv/h. JAIF detected that in the same condition, Dose in the exclusion area for Unit 1 was 3.855E-02 Sv/h. The dose calculations were performed for short-term individual effective dose and long-term individual effective dose for each radius. Collective dose was determined by assuming a single homogeneous population. Exposures in long-term individual doses were modeled through a cloudshine pathway and groundshine. While the calculations for the long-term doses were modeled through groundshine pathway (via the food chain). The potential exposure situation for a nuclear reactor facility was not a typical accident affected by the fuel in the core which releases radioactive material to the environment. Thus the effects to the public and types of consequences may be different.

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Table 4. Fission product source term.

Nuclide	Source term (Bq)			
	Unit-1	Unit-2	Unit-3	Total
Kr-85	1.34E+16	1.00E+16	8.35E+15	3.17E+16
Kr-85m	2.09E+17	1.55E+17	1.30E+17	4.94E+17
Kr-88	5.56E+17	4.11E+17	3.43E+17	1.31E+18
Xe-133	1.74E+18	1.29E+18	1.07E+18	4.10E+18
Xe-135	5.07E+17	3.75E+17	3.13E+17	1.19E+18
Sr-90	9.52E+12	7.05E+14	5.88E+14	1.30E+15
Ru-106	6.38E+13	4.71E+15	3.93E+15	8.70E+15
Te-132	1.25E+14	9.21E+15	7.68E+15	1.70E+16
I-131	4.45E+16	1.51E+17	1.26E+17	3.21E+17
I-132	6.37E+16	2.17E+17	1.81E+17	4.62E+17
I-133	8.93E+16	3.04E+17	2.54E+17	6.47E+17
I-135	8.33E+16	2.84E+17	2.37E+17	6.04E+17
Cs-134	2.16E+13	1.60E+15	1.33E+15	2.95E+15
Cs-137	1.44E+13	1.07E+15	8.90E+14	1.97E+15
Ba-140	1.52E+14	1.13E+16	9.38E+15	2.08E+16
Ce-144	1.15E+14	8.52E+15	7.10E+15	1.57E+16

Table 5. Effective dose and collective dose vs distance.

Distance (km)	Shorterm dose (Sv/h)			Longterm dose (Sv/y)	collective dose (man-Sv)
	Unit-1	Unit-2	Unit-3	Total	
0.80	3.46E-01	9.23E-01	7.68E-01	1.13E+02	4.09E+05
3.00	1.05E-01	2.79E-01	2.32E-01	4.36E+01	1.04E+06
5.00	5.23E-02	1.39E-01	1.16E-01	2.11E+01	3.58E+05
10.00	1.46E-02	3.88E-02	3.23E-02	8.11E+00	2.02E+06
15.00	4.68E-03	1.25E-02	1.04E-02	3.66E+00	1.86E+05
20.00	4.26E-03	1.14E-02	9.46E-03	3.05E+00	1.52E+06
30.00	4.14E-03	1.10E-02	9.17E-03	3.87E+00	3.68E+06
40.00	3.21E-03	8.55E-03	7.11E-03	4.37E+00	4.35E+06
50.00	3.08E-03	8.22E-03	6.84E-03	3.28E+00	5.19E+06
60.00	9.92E-04	2.64E-03	2.20E-03	1.73E+00	2.58E+06
70.00	9.92E-04	2.64E-03	2.20E-03	1.40E+00	3.10E+06
80.00	6.40E-04	1.71E-03	1.42E-03	8.38E-01	1.67E+06
100.00	3.13E-04	8.33E-04	6.94E-04	3.41E-01	3.01E+06
150.00	1.32E-04	3.53E-04	2.94E-04	8.45E-02	2.47E+06
200.00	1.01E-04	2.70E-04	2.25E-04	5.30E-02	9.35E+05
250.00	5.56E-05	1.48E-04	1.23E-04	3.75E-02	1.83E+06
300.00	4.28E-05	1.14E-04	9.50E-05	3.07E-02	8.13E+05
350.00	3.42E-05	9.12E-05	7.59E-05	2.59E-02	1.76E+06
400.00	2.78E-05	7.42E-05	6.17E-05	2.20E-02	7.76E+05
450.00	2.34E-05	6.22E-05	5.18E-05	1.92E-02	1.68E+06
500.00	1.93E-05	5.15E-05	4.29E-05	1.67E-02	7.37E+05
600.00	1.45E-05	3.87E-05	3.22E-05	1.32E-02	3.79E+06
700.00	1.13E-05	3.01E-05	2.50E-05	1.08E-02	6.65E+05
800.00	8.88E-06	2.37E-05	1.97E-05	8.89E-03	3.40E+06
900.00	7.23E-06	1.93E-05	1.60E-05	7.46E-03	5.93E+05

The dose calculations were performed for short-term individual effective dose and long-term individual effective dose for each radius. Collective dose was determined by assuming a single homogeneous population. Exposures in long-term individual doses were modeled through a cloudshine pathway and groundshine. While the calculations for the long-term doses were modeled through groundshine pathway (via the food chain). The potential exposure situation for a nuclear reactor facility was not a typical accident affected by the fuel in the core which releases radioactive material to the environment. Thus the effects to the public and types of consequences may be different. According to The Basic policy in the ICRP (2007) recommendation, in the emergency activities such as nuclear accident the annual dose to the people is allowed up to 100 mSv. It's defined as emergency exposure situation. Post emergency situation, the annual dose is limited within 20 mSv for the existing exposure situation for long-term exposure. And finally, additional exposure dose reduce less than 1 mSv/y should be achieved for longterm goal (Yoshida, 2013). The dose data in Table

5, indicates a short dose in the range of 24-48 hours after the accident and calculated for each unit. Dose in the exclusion area (within 800 m) for Unit 1 is 3.46E-01 Sv/h. JAIF detected that in the same condition, Dose in the exclusion area for Unit 1 was 3.855E-02 Sv/h. This difference may be came from the assumption. In this study, we assumed that the core damaged already reached 70%, while when JAIF detected, the core damaged was just 10%. And for Unit 2 and 3, the doses in the exclusion area are 9.23E-01 Sv/h and 7.68E-01 Sv/h, respectively. In the other hands, JAIF detected that the dose in the exclusion area were 4.00E-01 Sv/h and 4.00E-01 Sv/h respectively. The differences between JAIF results and current results may be caused the differences in meteorological conditions. Meteorological data used in this calculation may be incomplete compared to the actual condition.

Collective dose is influenced by individual dose and population density. To reduce the collective dose it can be achieved by reducing the exposure and/or reducing population density. Reduction of exposure can be done by decontamination to contaminated area, while reduction in population density can be done by relocating the population to other areas that are not contaminated. According to the ICRP (2007), collective dose limits is 20.000 man-Sv. Thus, for areas that exceed this limit are required to reduce the collective dose to be below the limit.

Determination of countermeasures are required to mitigate radiation exposure and health risks from radiation accident. The action of countermeasures can be divided into two types that are short-term countermeasures such as providing iodine pills, sheltering and evacuation, and long-term countermeasures which include population relocation and restriction to consume local contaminated food product. According to the criteria mentioned by the ICRP 63 (1992), evacuation is recommended whenever the effective dose exceeds 50 mSv. The zone for sheltering is recommended when the effective dose exceeds 10 mSv. By applying ICRP 63 (1992) rules and based on short dose calculation results in Table 5, then the sheltering actions required for the radius of 30 km and evacuation measures for distance less than 30 km.

According to the calculation results in Table 5, iodine pills delivery to residents were performed in the area within radius of 40 km; sheltering action is within the radius of 30-60 km for 6 months; evacuation is subjected for the area more than 30 km from reactor for 2 years; relocating is subjected to radius of 60 km for 3 months; decontaminations area within 20 km radius for 10 years; and food restriction on the radius of 30km for 5 years. If these actions are performed especially if comprehensive decontaminations are performed, then the countermeasures will diminish by the time in accordance with the progress of the decontamination process.

Table 6. Mean individual risk of incidence (Sv/y) vs distance.

Distance (km)	Total	lung	thyroid	bone marrow	breast	colon	skin
0.80	6.50E+00	5.94E+00	4.49E-01	2.72E-01	4.68E-01	1.08E+00	2.44E+00
3.00	2.57E+00	2.11E+00	2.17E-01	1.27E-01	2.28E-01	4.63E-01	8.34E-01
5.00	1.22E+00	9.14E-01	1.17E-01	6.70E-02	1.25E-01	2.33E-01	3.51E-01
10.00	4.33E-01	2.32E-01	5.92E-02	3.24E-02	6.36E-02	1.00E-01	7.37E-02
15.00	1.88E-01	7.87E-02	2.97E-02	1.60E-02	3.20E-02	4.74E-02	2.00E-02
20.00	1.49E-01	4.10E-02	2.77E-02	1.48E-02	2.99E-02	4.18E-02	2.87E-03
30.00	1.87E-01	4.59E-02	3.59E-02	1.91E-02	3.87E-02	5.36E-02	1.52E-03
40.00	2.16E-01	6.72E-02	3.86E-02	2.06E-02	4.18E-02	5.90E-02	1.50E-02
50.00	1.42E-01	4.03E-02	1.21E-01	1.26E-02	2.58E-02	3.61E-02	1.03E-02
60.00	8.11E-02	2.06E-02	2.86E-02	8.04E-03	1.63E-02	2.26E-02	1.25E-03
70.00	6.30E-02	1.60E-02	3.89E-02	5.97E-03	1.21E-02	1.68E-02	4.71E-02
80.00	3.67E-02	9.11E-03	2.82E-02	3.41E-03	6.92E-03	9.62E-03	3.70E-02
100.00	1.38E-02	3.19E-03	1.72E-02	1.20E-03	2.45E-03	3.39E-03	2.55E-02
150.00	2.47E-03	3.68E-04	9.05E-03	1.41E-04	2.98E-04	4.00E-04	1.50E-02
200.00	1.36E-03	1.48E-04	6.61E-03	5.66E-05	1.29E-04	1.62E-04	1.11E-02
250.00	8.66E-04	6.55E-05	5.18E-03	2.47E-05	5.58E-05	7.10E-05	8.79E-03
300.00	6.99E-04	4.91E-05	4.30E-03	1.86E-05	4.22E-05	5.34E-05	7.31E-03
350.00	5.83E-04	3.93E-05	3.64E-03	1.49E-05	3.40E-05	4.29E-05	6.22E-03
400.00	4.93E-04	3.22E-05	3.11E-03	1.23E-05	2.80E-05	3.52E-05	5.32E-03
450.00	4.28E-04	2.74E-05	2.72E-03	1.04E-05	2.38E-05	2.99E-05	4.67E-03
500.00	3.71E-04	2.31E-05	2.38E-03	8.83E-06	2.02E-05	2.53E-05	4.10E-03
600.00	2.93E-04	1.80E-05	1.89E-03	6.85E-06	1.58E-05	1.97E-05	3.27E-03
700.00	2.38E-04	1.45E-05	1.54E-03	5.51E-06	1.28E-05	1.58E-05	2.68E-03
800.00	1.97E-04	1.18E-05	1.27E-03	4.50E-06	1.05E-05	1.29E-05	2.22E-03
900.00	1.65E-04	9.93E-06	1.07E-03	3.78E-06	8.84E-06	1.09E-05	1.87E-03

JAIF reported (2011), regarding to the Fukushima Daiichi accidents, such countermeasures action that have been done are as follows :

1). Shall be evacuated for within 3km from NPS, Shall stay indoors for within 10km from NPS (issued at 21:23, Mar. 11th)

2). Shall be evacuated for within 10km from NPS (issued at 05:44, Mar. 12th)

3). Shall be evacuated for within 20km from NPS (issued at 18:25, Mar. 12th)

4) Shall stay indoors (issued at 11:00, Mar. 15th), Should consider leaving (issued at 11:30, Mar. 25th) for from 20km to 30km from NPS

5).The 20km evacuation zone around the Fukushima Daiichi NPS is to be expanded so as to include the area, whenever annual radiation exposure is expected to be above 20mSv. People in the expanded zone are ordered to evacuate within a month or so. People living in the 20 to 30km and other than the expanded evacuation area mentioned above, are asked to get prepared for going and staying indoors or evacuation in an emergency (issued on Apr14).

Countermeasures that have been done to the Fukushima Daiichi NPP are not different compared to the measures proposed in this study.

Risk Estimation

Radiological risk (including consequences) assessment for accident is required to determine the radiological risk (impact) of accidental discharges and then inform the requirements for accident management measures and emergency planning and preparedness. Risk models are given to the general population for these cancers. Table 6 shows the risk calculation. The greater of the probability (probability per year), the greater of consequences (individual or collective dose), besides, the radiological risk is even greater. For the same probability of different doses, the risk will be higher. According to the dose in Table 5, the radiological risk will diminish with increasing distance from reactor.

In December 2003, NSC issued "Interim Report on the Discussion of Safety Goals" to propose qualitative and quantitative safety goals to be applied consistently to all types of nuclear activities. "The average risk of early fatality for members of the public in the vicinity of the site boundary of a nuclear facility due to radiation exposure from nuclear accidents should not exceed approximately one in 1,000,000 a year. The average risk of cancer fatality for members of the public within a certain distance from a nuclear facility due to radiation exposure from nuclear accidents should not exceed approximately one in 1,000,000 a year."

Differences in the calculation for source term, consequences, countermeasures and risk at Fukushima Daiichi accident in this paper compare to the results already reported (NISA and NSC) could be caused by several things: 1) applied model for assessment, 2) the difference postulations and assumptions of accident (pessimistic or optimistic), and 3) the completeness of meteorological input data. But for some items, the difference was not so significant. Assessment performed in this paper could be a model to study nuclear power reactor under accident condition. Thus, it will be useful as an input for radiation protection, emergency preparedness, and in general for reactor safety and radiation safety.

Conclusion

The calculations of source term, consequences, and risks on Fukushima Daiichi accident are influenced by postulations, assumptions and assessment model. Instead of the limitations of meteorological data at the Fukushima Daiichi site, the calculation results have no significant difference compared to the calculations results performed by other researchers. The results of this assessments indicate that the assumptions to the calculation of consequences for the severe accidents simulations are more moderate than others assessment

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DISCUSSION

1. Dadong Iskandar – PTKMR BATAN :

Question : From where the source you obtain for Fukushima Daichi data?

Answer : The source that we use in my research was from the IAEA Report on September 2011.

2. Iin Kurnia PTKMR BATAN :

Question : Did you use primary or secondary data?

Answer : To calculate core inventory and source term, we use the primary data and for the postulation accident we use the secondary data.

Dose-Response Curve of Chromosome Aberrations in Human Lymphocytes Induced by Gamma-Rays *)

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Abstract. **Chromosome aberration is a biomarker to predict the level of cell damage caused by exposure to ionizing radiation on human body. Dicentric chromosome is a specific chromosome aberration caused by ionizing radiation and is used as a gold standard biodosimetry of individuals over exposed to ionizing radiation.In radiation accident the dicentric assays has been has been applied as biological dosimetry to estimate radiation absorbed dose and also to confirm the radiation dose received to radiation workers. The purpose of this study was to generate a dose resonse curve of chromosome aberration (dicentric) in human lymphocyte induced by gamma radiation. Peripheral blood samples from three non smoking healthy volunteers aged between 25-48 years old with informed consent were irradiated with dose between 0.1-4.0 Gy and a control using gamma teletherapy source. The culture procedure was conducted following the IAEA standard procedures with slight modifications. Analysis of dose-response curves used was LQ model $Y = a + \alpha D + \beta D^2$. The result showed that α and β value of the curve obtained were 0.021 ± 0.007 D and, 0.01 ± 0.003D^2, respectively. Dose response calibration curve for dicentric chromosome aberrations in human lymphocytes induced by gamma-radiation fitted to linier quadratiq *model.* In order to apply the dose response curve of chromosome aberration disentric for biodosimetry, this standar curve still need to be validated.**

Key words: chromosome aberrations, dicentric assay, standard calibration curve, biodosimetry.

*) The complete paper will be published in Atom Indonesia Journal

The Performance Assessment of TLD BARC System of PTKMR-BATAN to Support Radiation Protection Program in Indonesia

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Abstract. **Individual monitoring constitutes an integral part of radiation protection programme in Indonesia and all radiation workers have to be covered by individual monitoring. Since the last century, radiation protection philosophy has changed. With the indigenious development, thermoluminiscence (TLD) can substitute the role of film badge, because TLD offering better process in evaluation, faster and more economics, because of its reusable for 20 times of using. The dosimetric characteristics of the equipment and materials can be changed with the function of time and frequency of use, therefore the response or sensitivity of TLD should be checked. In addition, reduction of the dose limit makes increasing quality requirements on a particular individual monitoring. One of personnel dosimeters used in Indonesia is TLD BARC, made in India. This paper presented the performance of TLD BARC system in the year of 2011, 2012 and 2013, based on the criteria of ANSI and JCGM 100: 2008. In the last three years, the observation of the performance of TLD BARC system of PTKMR-BATAN was still in a good agreement with ANSI's criteria, however the uncertainties were fluctuated from 5 to 20%. Hopefully the QA Program can support radiation protection program in Indonesia because PTKMR-BATAN provides dose evaluation services for more than 28,000 radiation workers in Indonesia in the year of 2013.**

Keywords: Performance, TLD BARC System, radiation protection.

Introduction

The dosimetric characteristics of the equipment and materials can be changed with the function of time and frequency of use, therefore response or sensitivity of the equipment should be checked before using (IAEA Safety Standard Series, 1999; Pradhan et al., 2002; IAEA-TECDOC, 1999) so do TLD's sensitivity. Monitoring services at any laboratory need access to calibration facilities. Services must be able to provide radiation field needed to evaluate the performance of the dosimetry system. Secondary Standard Instrument should be available to measure radiation beam quantity, required by external dose monitoring services and measuring equipment as well as the radiation source must be calibrated and traceable to national standards or to the Secondary Standard Laboratory (SSDL) or Primary Standard Dosimetry Laboratory (PSDL) (SRS-IAEA16, 2000).

In addition, radiation protection philosophy has changed since the last century. Radiation protection begins by limiting exposure at a certain level, where the stochastic effects of radiation hazards are ignored. Radiation dose assessment becomes more important when the risk due to the radiation dose producing stochastic effects, for example, radiation can induce carsinogenetic without threshold. Currently the recommended annual dose limit based on the results of the epidemiological investigation, radiobiologic and health risks of exposure due to ionizing radiation and opinions expressed on additional health risks, which may be received by workers and communities. With the advancement of science and technology in the field, such restrictions gradually reduced and the International Commission on Radiological Protection (ICRP-60) and PERKA BAPETEN No.4/2013 (BAPETEN, 2013) has reduced the dose limit of 50 to 20 mSv, on average per year for radiation workers and from 5 to 1 mSv for the public. Reduction of the dose limit makes increasing quality requirements on a particular individual monitoring: accuracy, performance and recording level. Based on ICRP, Recording Level Value (RLV) for monthly service has been revised from 0.42 mSv (ICRP-26/177) to be 0.17 mSv (ICRP-35/1982), and since (ICRP-75/1999) RLV revised be 0.085 mSv for worker exposure. Consequently dosimeter test requirements become more stringent (ICRP, 1999).

To implement the ICRP, PTKMR-BATAN carried out Quality Assurance Program of TLD BARC system, which consists of internal and external Quality Assurance (QA). Internal QA includes periodically checking EHT and Light source, physical verification, and annealing oven, test the linearity and stability of the TLD system. External quality assurance program carried out by following the intercomparison, held by SSDL- Jakarta [7], appendix

1. This paper presents the performance assessment of TLD BARC system in the year of 2011, 2012 and 2013, which includes test of EHT, light source, linearity and sensitivity or response of TLD BARC system, based on ANSI's Criteria and JCGM 100: 2008.

Materials and Methods

Algorithms for dose evaluation (JCGM 100, 2008)

There are 3 disks in TLD BARC Card: D_1, D_2, and D_3 (Figure 1 or Figure 2). D_1: is the dosimeter readings under the filter Cu-Al minus control card reading, D_2 : is dosimeter readings under plastic window - minus control card reading, and D_3 : the dosimeter readings below the open window – control card reading. D_1, D_2, and D_3 = 0, if it is smaller than the RV (Reported Value). RV = 0.2 mSv for gamma, and 0.5 mSv for beta and 0.05 mSv, for low energy X-rays.

Experiment and analysis

The Equipment used in this experiment is TLD BARC Card (Figure 1), calibrated oven, ^{137}Cs calibrator, TLD Reader Model TL 1010 and computer systems (Figure 3). TLD BARC consists of Teflon disk and TLD (CaSO4: Dy) (Figure 1), with a ratio of CaSO4: Dy and Teflon 1: 3. Z_{eff}: 15.1 and TLD disk density: 2.52 g/cm^3. Softening Point of Teflon: 330 ^oC and Main Glow Peak Temperature: 230^oC (Figure 4), with a sensitivity of TLD discs: 30-40 x of LiF TLD-100. Fading: 2-3% in 6 months. The influence of climate and sunlight can be ignored when covered with paper / poliethilen and inserted into the badge. Linear dose range: 0.1 to 20 mSv (within ± 10%).

Reusability was 20x. Energy response was TLD CaSO4: Dy Te disk that depends on the photon energy, especially at energies 30 keV to 200 keV, while above 200 kev energy is relatively flat. Dimensions TLD Card to card from Al: 52.5 mm x 30.0 mm x 1.0 mm, diameter of the hole in the plate Al: 12 mm and diameter disk TLD: 13.3 mm (Figure 2).

The algorithm used depends on the nature of the radiation. For Doses of gamma rays, if the ratio of D_3 to D_1 <1.3, only the gamma dose to be evaluated using the value of D_1, ie gamma dose = D_1 (or = 0 if less than RV).

TLD Reader model TL 1010 (Figure 3) is a semiautomatic, heated electrically, using a power supply 220-230 V, 50 Hz, with a linear heating profile. Heating method using nitrogen gas, with clamping temperature (300 ± 2) °C, in 10-12 seconds, using a light detector, with the PMT readout time of 100 seconds per badge. TLD reader can be used to read TLD with exposure range from 0.05 mSv - 2 Sv. Using IBM pC 486 or higher that the FDD and HDD, with SVGA myths, serial and parallel ports, Dos and MS Windows 95/98/NT 6:20. Accuracy was ± 30% reading at 1 mSv, and ± 10% above 10 mSv.

Checking the sensitivity of TLD Card should be done after 5 x readings to check for significant reduction of the TL sensitivity, in addition to verify the average sensitivity variations between batches of different batches. To do this, at least 25 Cards are irradiated of each batch with a known dose. If the average sensitivity of each batch was reduced by more than 15% compared to the reference card, then the batch should be withdrawn from service. If the variation within a batch sensitivity ≥20%, then the whole batch must be removed from service. Or all cards are individually tested to give exposure to the same dose, and the only card that shows the variation in sensitivity of 15% selected for further use. It should be noted in the logbook.

Linearity and Performance Test of TLD BARC

TLD BARC was annealed using a calibrated oven and be checked every month by using a thermocouple or thermometer system. Uncertainty's oven should be around ± 2 ^oC. The oven should not be used for any other purpose, such as for heating lunch boxes or organic materials. Each annealing was no more than 500 TLD cards. Conformity can be checked by reading the annealing annealed 5 cards with calibrated reader, readings should be as close as possible to the background, (50 ± 10) μSv.

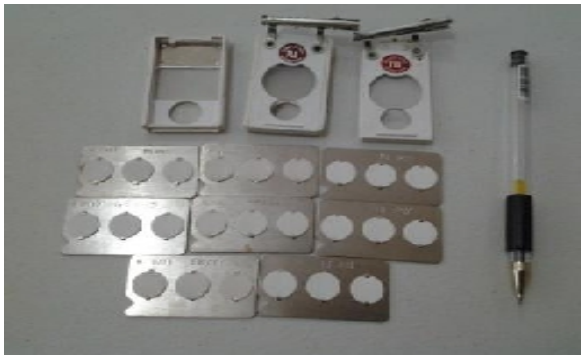

Figure 1. TLD BARC Card

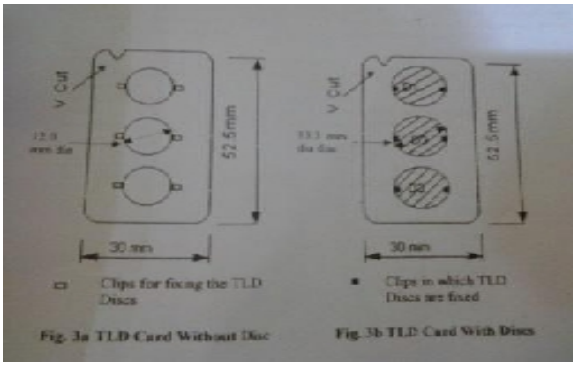

Figure 2. Assembly of TLD BARC

Figure 3. TLD Reader model TL 1010

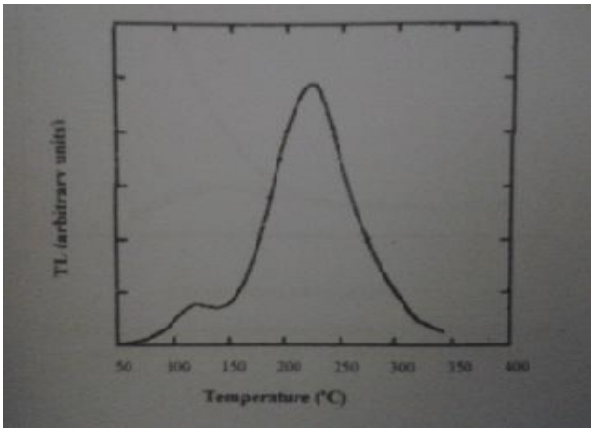

Figure 4. Glow Curve of TLD

Other controlling of TLD BARC reader TL 1010 is the EHT and Light Source controlling. TLD Card and tray are cleaned with acetone and dried at room temperature for 12-16 hours. Tray containing a TLD placed on the oven and the temperature was raised from room temperature up to 230 ° C, then kept / maintained for 4 hours. TLD card tray and be removed from the oven when the temperature is below than 80 ^{o}C.

TLD Card were irradiated in a known gamma radiation field ^{137}Cs. TLD Card was put on the surface of PMMA phantom, size: 30 cm x 30 cm x 30 cm at the distance of source of 200 cm at the height 1m from the floor. In this experiment, TLD Cards were irradiated at 9 to 11 variation of doses from 0.1 to 20 mSv. Irradiation of TLD Card was done in the year of 2011, 2012 and 2013. After 24 hours irradiated, TLD Card were read using TLD Reader semi automatic model TL 1010.

ANSI criteria for testing the performance of Personnel Dosimeters (LMRN-PTKMR, 2011).

ANSI (American National Standard Institute) provides a procedure for performance testing of dosimetry system. These procedures are periodically reviewed by the ANSI. According to the ANSI's criteria, performance assessment, P_i or bias, B_i of Dosimeter TLD system can be determined by the following equation:

$$P_i = B_i = \frac{H_M - H_T}{H_T} \qquad (1)$$

Where H_T is true dose (mSv) and H_M is a measured dose (mSv).

B shows the bias, if B is positive, then reported dose greater than the dose given, otherwise if B is negative, the reported dose are less. Standard deviation, S (σ_p).

$$S(\sigma_p) = \left[\frac{\sum_{i=1}^{n}(Pi - \overline{P})^2}{n-1}\right]^{1/2} \qquad (2)$$

If the S is large, it indicate a lack of precision or a large random scatter. Performance is acceptable if the level of tolerance, L not be exceeded by the sum of absolute values B and standard deviation, S. $[B] + \sigma_p \leq L$. Tolerance levels, L is a quantity which collectively reflects the overall uncertainty in the calibration, measurement and evaluation of dose and it takes into account the recommendations of the Competent Authorities in the field of Radiation Protection.

Upper limits of the permissible accuracy limit is upper part of the trumpet curve, is given by:

$$H_{UL} = 1{,}5\left[1 + \frac{H_o}{2H_o + H_1}\right] \qquad (3)$$

Lower limits of accuracy are allowed upper limit of the trumpet curve.

$$H_{LL} = \frac{1}{1{,}5}\left[1 - \frac{2H_o}{H_o + H_1}\right] \text{ for } H_1 \geq H_o \qquad (4)$$

$$H_{LL} = 0 \text{ for } H_1 < H_o \qquad (5)$$

Where H_1 is true dose (mSv) and H_o is the lowest measurable dose (recording level). H_o can be calculated from this equation =1mSv x frequency of service, divided by 12. For the monthly period, H_o = 0.085 mSv according to ICRP 60/75 (Pradhan et al., 2002).

Criteria JCGM 100:2008 to test the Performance Personal Dosimeter (JCGM 100, 2008)

Overall uncertainty of the dosimeter system is determined from a combination of Type A and Type B. Uncertainty type A identified by standard deviation, σ of a series of measurements. Type A of uncertainty can be reduced by increasing the number of measurements. Type A uncertainty is obtained from direct measurement and type B uncertainties derived from previous measurements, or from reference. Type A uncertainties include: statistical uncertainty, u_A, uncertainty of resolution, u_{res}, and performance uncertainty, u_p. Whereas type B uncertainties include: uncertainties of irradiating TLD, u_{irr}, and uncertainty of fading, $u_{fad.}$

Combination of uncertainty, u_c

$$u_c = \sqrt{u_A^{\,2} + U_B^{\,2}} \qquad (6)$$

where expanded Uncertainty, u_{exp}:

$$u_{exp} = k \times u_c \qquad (7)$$

where k is coverage factor, depend on the $v_{eff,}$ where v_{eff} is effective degrees of freedom, obtained from the Welch-Satterthwaite.

$$v_{eff} = \frac{u_c^{\,4}(y)}{\sum_{i=1}^{n} \frac{u_i^{\,4}}{v_i}} \qquad (8)$$

Value of k can be obtained from the t-distribution, or simply, k is 2 for 95% confidence level.

Results and Discussion

The TLD BARC Reader (TL 1010) gives the information of EHT and Light Source before reading the TLD BARC so we know the stability of the reader. In this experiment we only presented the stability of EHT from 4 July to 24 July 2013 and the Light Source from 27 May to 1 July 2013. Control Chart of EHT and Light source of the Reader were presented in Figure 5 and 6. The performance of TLD reader was in a good agreement, in the range of two sigma (between LWL and UWL).

Observation of the performance of TLD BARC system was conducted in the year of 2011, 2012 and 2013. The results are presented in Figure 7 to Figure 12 and Table 1 to Table 3. The BARC TLD system performance was in a good agreement with ANSI

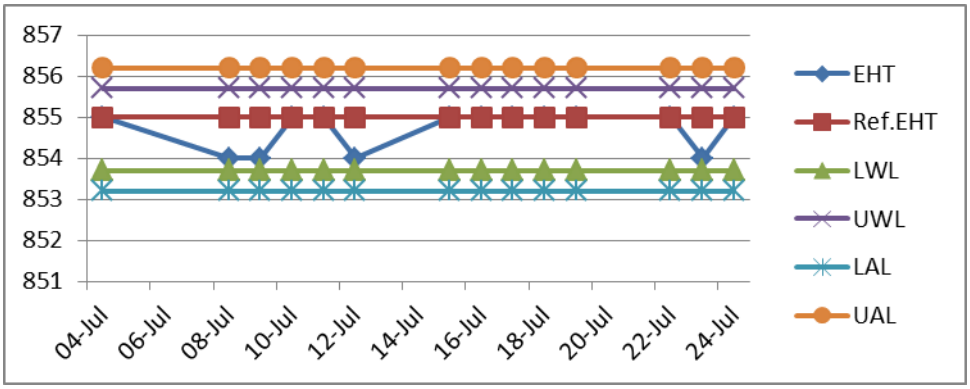

Figure 5. Control Chart of EHT of the TLD BARC Reader TL 1010.

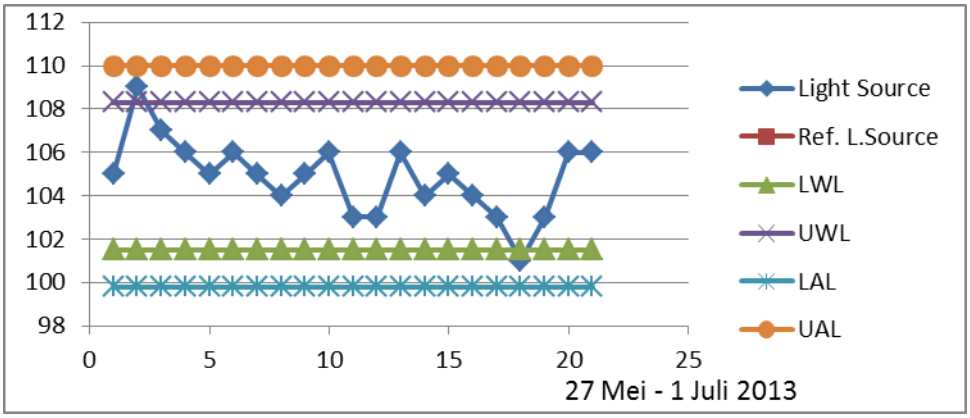

Figure 6. Control Chart of Light Source of the TLD BARC Reader TL 1010.

(still within the trumpet curve) however the uncertainty varies from 5 to 20%, with a confidence level of 95%.

The sensitivity of TLD BARC is still in accordance with the Manufacturer Specifications Accuracy ± 30% reading at 1 mSv, and ± 10% above 10 mSv. The response of TLD BARC system was presented in Figure 13 and 14. The response of TLD BARC in the year of 2012 and 2013 was larger when compared with 2011. This may be influenced by an environmental condition, annealing, frequency of using and aging on the TLD system.

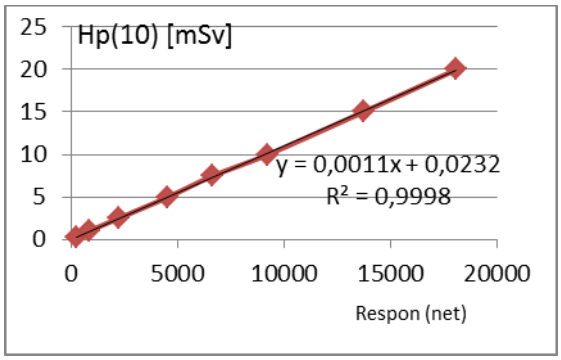

Figure 7. Performance of TLD BARC System (2011)

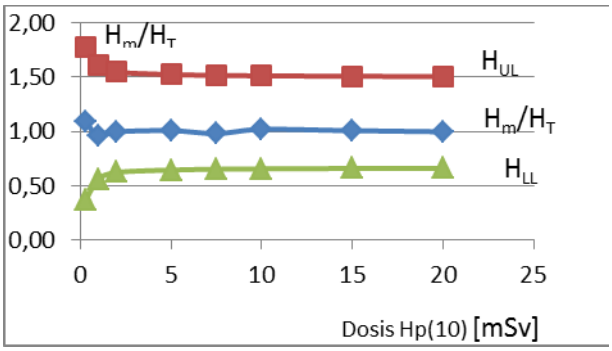

Figure 8. Performance of TLD BARC system (2011) based on ANSI criteria

Table 1. Performance of TLD BARC system (2011) based on JCGM 100:2008.

No.	Component	Uncertainty (%)
1	$u_{irradiation}$	2
2	$u_{Performance}$	1
3	$u_{statistic}$	1
4	$u_{resolution}$	0.2
5	u_{fading}	0.5
	U_{exp} (CL=95%)	5

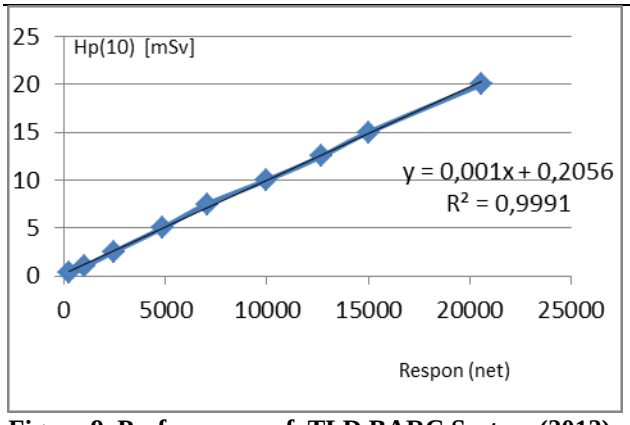

Figure 9. Performance of TLD BARC System (2012).

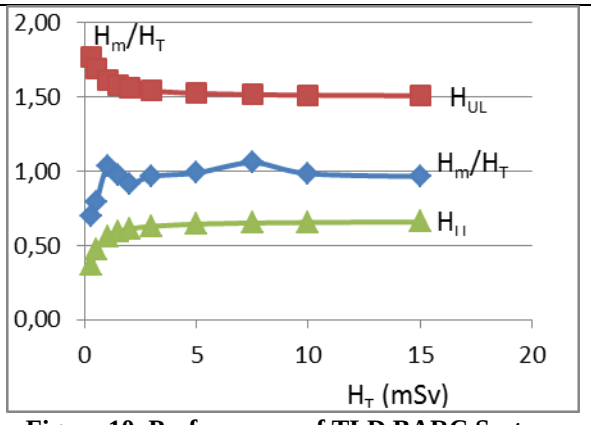

Figure 10. Performance of TLD BARC System (2012) based on ANSI criteria.

Table 2. The Performance of TLD BARC system (2012) based on JCGM 100:2008.

No.	Component	Uncertainty (%)
1	$u_{irradiation}$	2
2	$u_{Performance}$	9
3	$u_{statistic}$	4
4	$u_{resolution}$	0.2
5	u_{fading}	0.5
	U_{exp} (CL=95%)	20

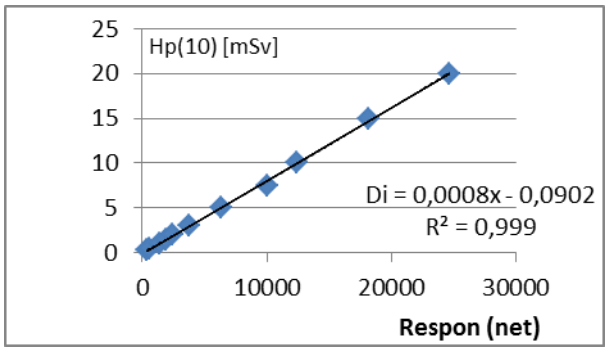

Figure 11. Performance of TLD BARC System (2013)

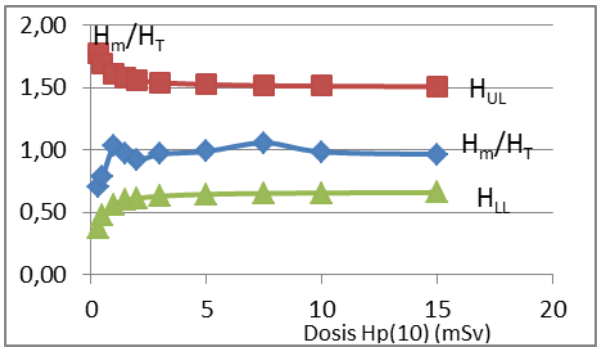

Figure 12. Performance of TLD BARC system (2013) based on ANSI criteria

Table 3. Performance of TLD BARC system (2013) based on JCGM 100:2008

No.	Component	Uncertainty (%)
1	$u_{irradiation}$	2
2	$u_{Performance}$	6
3	$u_{statistic}$	2.4
4	$u_{resolution}$	0.03
5	u_{fading}	0.5
	U_{exp} (CL=95%)	13.5

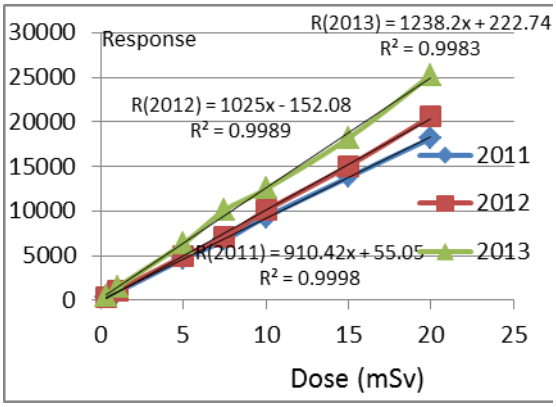

Figure 13. The response of TLD BARC system

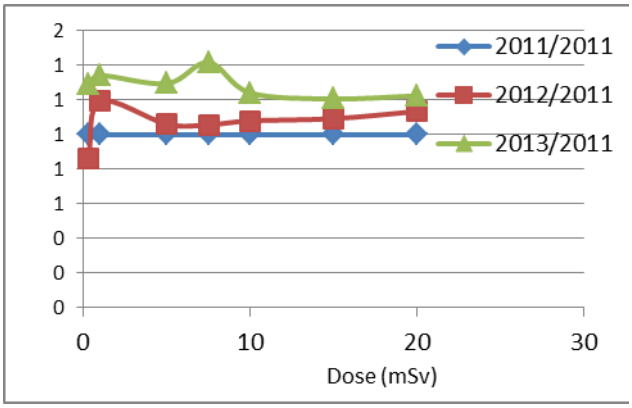

Figure 14. The response of TLD BARC system

2011-2013.

Radiation Protection Programme

The effectiveness of the radiation protection programme relies on the implementation of various components, including the adoption of QA programs (Quality Assurance). Expansion of the QA program should be consistent with the number of workers who will be monitored and the amount of exposure expected in the workplace and by implementing the monitoring programme. Based on the Activity Report of PTKMR BATAN, in the period of January to June 2013, there are 13,900 radiation wokers has been evaluated by sub division of Radiation Protection, Safety and Health, PTKMR-BATAN. So in the year of 2013, a number of radiation workers will be evaluated by PTKMR-BATANis about 28,000 workers.

Everyone who involved in the evaluation program of external exposure is responsible for the quality and therefore he/she has to implement the QA program and QC procedures. General requirements relating to QA radiation worker exposure presented in the Basic Safety Standard (BSS) (JCGM 100, 2008) and general instructions given in the Safety Guide as well as ISO / IEC Guide 17025 is widely used by regulatory agency to accredit testing and calibration program (ISO-IEC:17025, 2008).

Conclusions

a. The performance of TLD BARC system in the year of 2011 to 2013 was in a good agreement with ANSI criteria (still in the trumpet curve).
b. The uncertainty of TLD BARC system in the past of three years are varied from 5 to 20 %, with a confidence level of 95%,
c. The Performance of TLD BARC is still in accordance with the Manufacturer Specifications

2011-2013 compared with 2011.

Accuracyof ± 30% reading at 1 mSv, and ± 10% above 10 mSv.

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DISCUSSION

Iin Kurnia, PTKMR BATAN :

Questions:

Did you use primary or secondary data for your research?

Answer:

We used primary data. TLD card was irradiated in a known gamma radiation field, with 9-11 variation of doses, and they were read by TLD reader. The performance of TLD read was assessed by using ANSI and trumpet curve and the uncertainty was calculated by JCGM 200:2008.

Appendix 1. Quality Audit Report of Hp(10), 2011

LABORATORIUM METROLOGI RADIASI NASIONAL
PUSAT TEKNOLOGI KESELAMATAN DAN METROLOGI RADIASI
BADAN TENAGA NUKLIR NASIONAL
Jalan Lebak Bulus Raya No. 49, Pasar Jumat
Kotak Pos 7043 JKSKL, Jakarta Selatan 12070, Indonesia
Telp.(021) 7513906, Faks (021) 7657950, E-mail:ssdl.jakarta@batan.go.id

LAPORAN AUDIT MUTU PENENTUAN DOSIS PERORANGAN *Hp (10)*

No. LMR-N/KAL/QA/XII-11/09

Institusi	:	Laboratorium PTKMR BATAN	Dosimeter	:	Termoluminisensi (TLD)
Alamat	:	Jl. Lebak Bulus Raya No. 49, Jakarta 12440	Tgl. Penyinaran LMR-N	:	22 Oktober 2011
No. Telp.	:	021 – 7513906	Tgl. Proses	:	27 Oktober 2011
No. Fax.	:	021 – 7657950			

HASIL :

Kode Dosimeter	*Hp(10)* yang diberikan oleh LMR-N * (mSv)	*Hp(10)* yang dinyatakan oleh Lab. Peserta (mSv)	Nisbah Lab. Peserta / LMR-N	Nisbah yang dapat diterima **
BARC/ 1.1	0,2	0,27	1,35	
BARC/ 1.2		0,24	1,20	Batas atas : **2,00**
BARC/ 1.3		0,26	1,30	Batas bawah : **0,00**
BARC/ 1.4		0,25	1,25	
BARC/ 2.1	0,5	0,64	1,28	
BARC/ 2.2		0,56	1,12	Batas atas : **1,83**
BARC/ 2.3		0,66	1,32	Batas bawah : **0,29**
BARC/ 2.4		0,63	1,26	
BARC/ 3.1	0,9	1,11	1,23	
BARC/ 3.2		1,08	1,20	Batas atas : **1,73**
BARC/ 3.3		1,09	1,21	Batas bawah : **0,42**
BARC/ 3.4		1,08	1,20	
BARC/ 4.1	1,4	1,71	1,22	
BARC/ 4.2		1,64	1,17	Batas atas : **1,67**
BARC/ 4.3		1,75	1,25	Batas bawah : **0,50**
BARC/ 4.4		1,47	1,05	

*) Ketidakpastian standar gabungan relatif untuk penyinaran dosimeter termoluminisensi (TLD) adalah 4,9 % untuk tingkat kepercayaan 2σ (95%).

**) Nisbah yang dapat diterima didasarkan pada Gambar. 2 dari pedoman IAEA : *Assessment of Occupational Exposure due to External Sources of Radiation* (RS-G-1.3, 1999)

Tanggal 7 Desember 2011

Dra. C. Tuti Budiantari
Ketua Penyelenggara

Prof. Eri Hiswara MSc.
Manajer Kalibrasi

Presentation of Plenary Session I
10 October 2013

Presentation of Panel Discussion I
10 October 2013

Presentation of Plenary Session II
11 October 2013

Presentation of Panel Discussion II
11 October 2013

Topic A :
Exposures from Natural and/ or Man-Made Radiation

Topic B :

Occupational and Medical Radiation Exposures

Topic C :
Health and Environmental Effects of Radiation

Topic D :

Radiological/Nuclear Emergency Preparedness and Response