ISSN 1412-3258

2020 ANNUAL NUCLEAR SAFETY SEMINAR PROCEEDING

SKN 2020

INNOVATIONS TO SUPPORT NUCLEAR SAFETY AND SECURITY FOR ADVANCED HUMAN RESOURCES AND EXCELLENT INDONESIA

BADAN PENGAWAS TENAGA NUKLIR
BAPETEN

UNIVERSITAS INDONESIA

26 OCTOBER 2020
JAKARTA

ISSN 1412-3258

2020 ANNUAL NUCLEAR SAFETY SEMINAR PROCEEDING

SKN 2020

INNOVATIONS TO SUPPORT NUCLEAR SAFETY AND SECURITY FOR ADVANCED HUMAN RESOURCES AND EXCELLENT INDONESIA

BADAN PENGAWAS TENAGA NUKLIR
BAPETEN

UNIVERSITAS INDONESIA

26 OCTOBER 2020
JAKARTA

TABLE OF CONTENTS

OPENING REMARKS

Assalamu alaikum warahmatullahi wabarakatuh,

With Allah's Almighty grace, the Nuclear Energy Regulatory Agency (BAPETEN) held 20th Annual Nuclear Safety Seminar on October 26, 2020 in a series of commemoration of the 25th National Technology Awakening Day (HAKTEKNAS XXV), with the theme:

"Innovation to Support Nuclear Safety and Security for Advanced Human Resources and Excellent Indonesia"

This theme is close to the theme of HAKTEKNAS XXV which carries "Innovation as a Solution" with the sub-theme of excellence in research and innovation to increase the independence of the Indonesian nation. It is to be grateful that the Minister of Research and Technology was pleased to be present to open and give a speech at this seminar. On several previous occasions, he invited the success of Indonesia's transformation from a natural resource-based country to an innovation-based country.

In accordance with the theme, the Nuclear Safety Seminar is expected to become a forum for scientific meetings between regulators and users, experts, and the public member through the exchange of information, knowledge, experience, and views to improve safety and security in the use of nuclear energy in Indonesia, linked to research and innovation excellence technology, for the nation's independence. The hope is that an independent, advanced, and prosperous Indonesia can be realized together.

In 2020, this seminar will be held in collaboration between BAPETEN and the Faculty of Mathematics and Natural Sciences - Universitas Indonesia with all its positive considerations.

In this seminar, 92 papers were submitted to the committee. After the assessment was held by the reviewer team, it was decided that as many as 62 papers could be presented at the seminar, consisting of 36 papers presented at the oral session and 26 papers presented in the form of short presentations. 3 (three) keynote speakers from BAPETEN, IAEA (International Atomic Energy Agency), and the Universitas Indonesia, will present material in accordance with the theme of this seminar and touch on current conditions that are still shrouded by the COVID-19 pandemic.

We would like to thank the officials within BAPETEN and the Universitas Indonesia and their staff, speakers who have delivered very useful materials, and speakers who have participated in this event, which in its implementation was held in the form of online seminars or webinars.

At the end of the word, I would like to thank all the committees who have done their best to hold this event and are committed to the success of the 2020 Nuclear Safety Seminar.

We, as the organizing committee apologize, if there are deficiencies in the implementation of this event.

Wassalamu alaikum warahmatullahi wabarakatuh

Jakarta, October 2020

Dr. Ir. Yudi Pramono, M.Eng.
Chairman of the Committee.

EDITORS AND COMMITTEES

This page is intentionally left blank…

HTGR Thermohydraulic Study on Steady-State Conditions Using ANSYS FLUENT

Agus Waluyo[1, a]) and Azizul Khakim[2, b])

[1,2] *Badan Pengawas Tenaga Nuklir*
Jalan Gadjah Mada no.8 Jakarta Pusat

[a)] a.waluyo@bapeten.go.id
[b)] a.khakim@bapeten.go.id

Agus Waluyo is the main author of this paper, and Azizul Khakim is the supporting contributor

Abstract: HTGR is a reactor type that uses helium gas as its coolant that has a high operating temperature. Many countries are considering building HTGR because it has higher thermal efficiency compared to PWR or BWR. Many studies have been carried out related to HTGR, one of them is related to thermohydraulic performance. Because the thermohydraulic performance is closely related to its safety, therefore this study will be carried out to determine the temperature distribution on the HTGR core, and the graphite reflector temperature will also be compared to experimental data. The HTGR type which is used as a reference in this research is the HTR-10 which has been built in China. The calculation of temperature distribution on HTGR core was carried out using ANSYS FLUENT code applying a porous medium approach and a 2-dimensional model with axisymmetry. The 2-D was applied to simplify the geometrical model as the HTR-10 core was a cylinder, and consequently, it would also speed up the CFD calculation and reach convergence. From the CFD calculations, the results showed that the core outlet helium gas temperature was 795 ^{0}C. As for the temperature measurement in the side reflector, the calculation results at the height of 80 cm produced the least difference compared to the experimental results, which is 4% (at a radius of 193 cm). And the biggest difference was 22% (at a radius of 93 cm). As for the height of 170 cm, the least difference with the experimental data was 4,3% (at a radius of 93 cm) and the greatest one was 12,76% (at 189 cm).

Keyword: HTGR, HTR-10, thermohydraulic, CFD, porous medium, FLUENT

INTRODUCTION

HTGR is a type of nuclear power plant which has a high operating temperature and uses helium as a coolant. Many countries are considering building HTGR because HTGR has higher thermal efficiency compared to PWR and BWR. Besides having high thermal efficiency, HTGR also has several advantages, for example, HTGR has a negative temperature reactivity, so that, when reactor temperature suddenly rises, the chain fission reaction will drop immediately, thus causing the reactor temperature return to its original temperature and also other advantages HTGR is the fuel of HTGR will not melt in severe accidents because it is made from graphite.

The HTGR reactor used in this research is the HTR-10 that has been built and operated in China. HTR-10 has a thermal power of 10 MW and has fuel in the form of pebble beds. The coolant used in the HTR-10 reactor is Helium because Helium is an inert gas that does not easily react with other elements. In the 2000s. The IAEA has initiated a Coordinated Research Project (CRP) with several countries to analyze the reliability of passive safety systems and the inherent safety of the HTR-10. The CRP benchmarked HTR-10 in steady-state to get the temperature distribution on the core of Full Power Initial Core (FPIC). The benchmarking analysis is carried out with codes and codes to experiment. The analysis was followed by several countries such as China, France, Indonesia, Japan, Netherlands, South Korea, Russia, South Africa, Turkey, United Kingdom, and United States. One of the studies that have been conducted is a benchmark analysis conducted by the UK using the WIMSTER Code program. The results of the study can be seen in **FIGURE 1** and **FIGURE 2** In the experimental analysis, the HTR-10 temperature measurements were made at certain points which can be seen in **TABLE 1.** [1]

TABLE 1. Measurement point on the side reflector of the HTR-10

No	R(cm)	Z(cm)	No	R(cm)	Z(cm)
1	193	80	13	60	-40
2	189	80			
3	167	80	14	40	234
4	133	80	15	60	234
5	117	80			
6	93	80	16	70	440
7	193	170	17	50	400
8	189	170	18	50	370
9	167	170	19	50	340
10	133	170			
11	117	170	20	26	340
12	93	170	21	26	300
			22	26	260

Many studies have been carried out related to HTGR, one of them is related to thermohydraulic performance. Because the thermohydraulic performance is closely related to its safety, therefore the purpose of this study is to determine the temperature distribution on the HTGR core and on the graphite reflector. The result will also be compared to experimental data. The temperature distribution of the HTR-10 reflector is very important to know because it can determine the strength of the reflector in receiving heat from the reactor core.

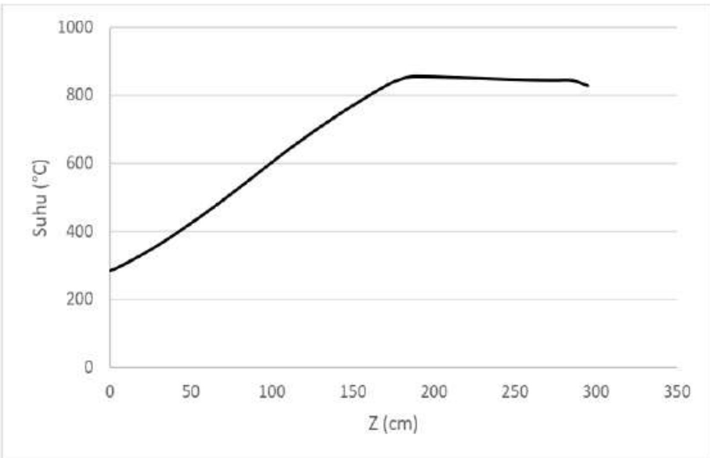


FIGURE 1. Results of the Axial Temperature Distribution at R = 0 cm with the WIMSTER code [2]

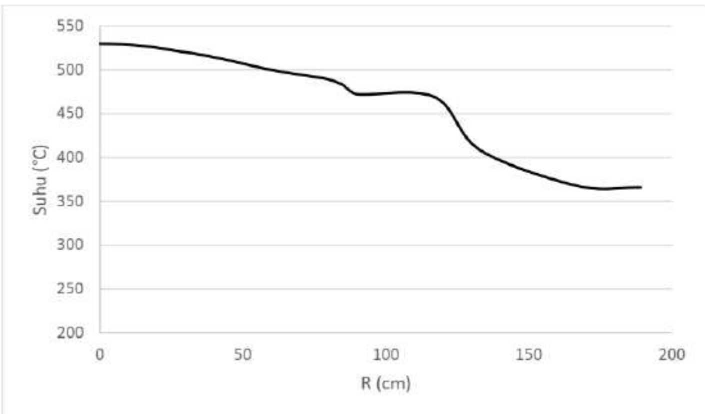


FIGURE 2. Results of Radial Temperature Distribution at Z = 80 cm with WIMSTER code [2]

DESIGN OF HTR-10

The HTR-10 reactor consists of pressure vessels, graphite, carbon brick, metal components, fuel, control rods, drive mechanism, small ball absorber, loading, and removal fuel systems. The active core of HTR-10 is surrounded by reflector graphite. The graphite reflector itself is categorized as a top reflector, side reflector, and bottom reflector. The lower part of the reactor core has a conical shape, and the lower part of the reactor core is joined by a tube that functions to remove fuel from the core. The cross-section of the HTR-10 reactor structure is shown in **FIGURE 3.**

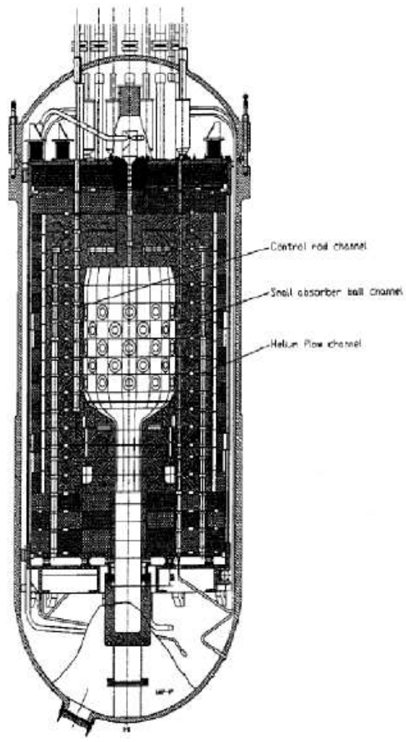


FIGURE 3. Cross-section of the HTR-10 reactor structure [2]

The side reflector is divided into two-part, first is graphite on the inside and second is carbon brick on the outside. Graphite has functioned as a neutron reflector for active core and carbon brick has a function as a thermal insulator and a neutron absorber. Side reflector has 20 channels close to the active core, 20 channels consisting of 10 channels for control rod channels, 7 channels for absorber ball, and 3 channels are provided for irradiation channels. The side reflector also has 20 helium cooling channels on the outside. The horizontal cross-section showing the position of the channels for control rods and helium cooling channels is shown in **FIGURE 4** and for the vertical cross-section shown in **FIGURE 5**.

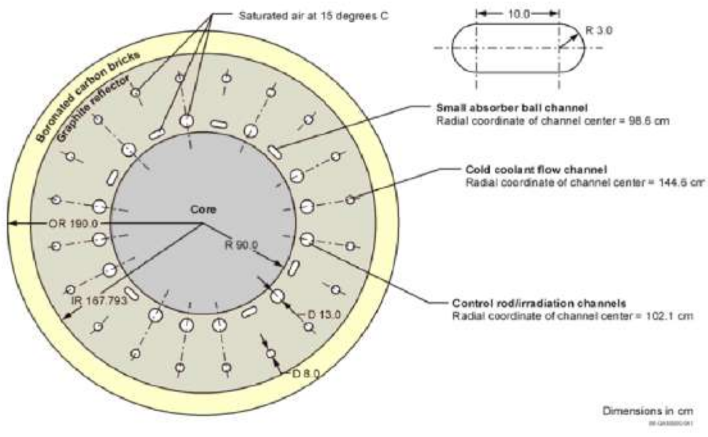


FIGURE 4. Horizontal cross-section of HTR-10

ISSN: 1412-3258 © 2020 BAPETEN – FMIPA UI

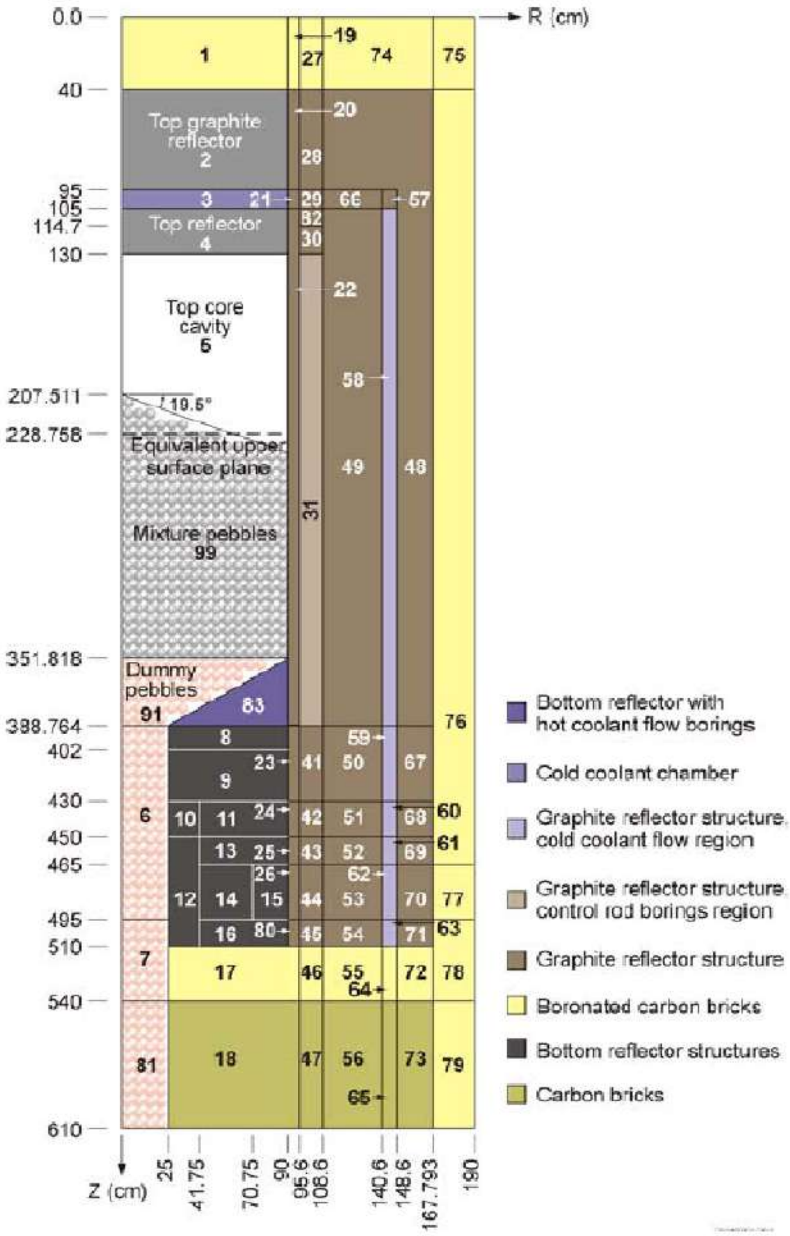


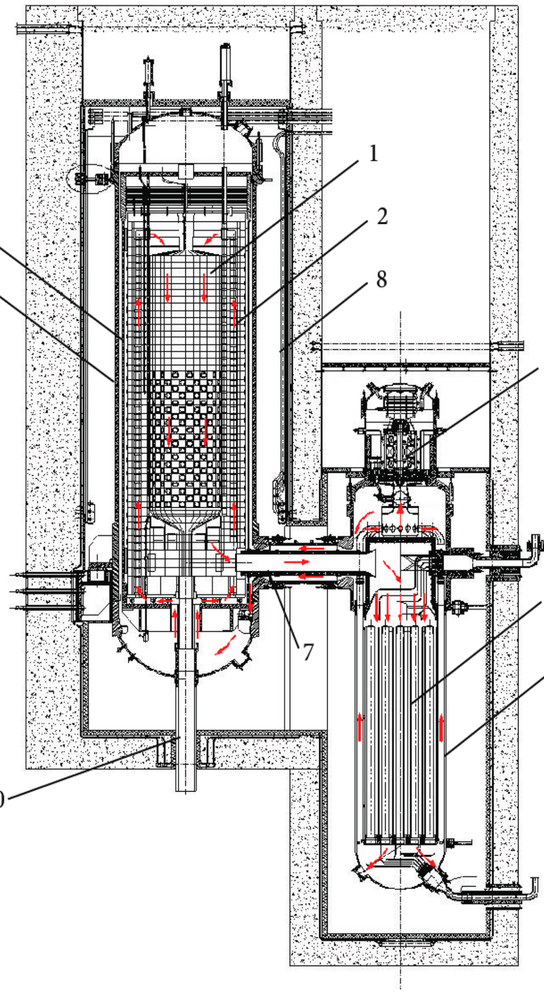
FIGURE 5. Vertical cross-section of HTR-10 [2]

HTR-10 uses helium gas as a coolant. Helium is pumped into an RPV with a helium circulator. After entering the RPV, helium flow drops through the annular space between the core vessel and the RPV. This flow changes to the top after passing through the bottom of the RPV. A small portion of helium flow enters the discharging tube and becomes one again with helium flow from the core which cools the fuel. Almost all helium goes to the support structure under the reactor core and enters the cold helium canal in the graphite side of the reflector block.

In the upper reflector, helium is collected in the cold plenum where is located at the top of the upper reflector. Some of the helium goes to the control rod canal which acts as a cooler for control rods. Starting at the top of the core, most of the helium goes down trough the reactor core to the bottom reflector. Finally, Helium goes to the hot helium plenum where located under the reactor core. Helium in hot plenum has a temperature of around 700 ^{0}C. From the hot plenum, Helium goes to RPV (reactor pressure vessel). Data of Helium distribution can be seen in **TABLE 2** and the flow scheme for Helium in the core can be seen in **FIGURE 6.**

TABLE 2. Helium distribution flow in HTR-10 Core [2] [3]

Distribution of helium flow	Percentage
Fuel discharging tube	1%
Control rods canal dan *small ball absorber*	2%
The gap between the graphite element	10%
Reactor core	87%



1. Reactor core
2. side reflector
3. *core barrel*
4. reactor pressure vessel
5. steam generator
6. steam generator vessel
7. *coaxial gas duct*
8. water-cooling panels
9. blower
10. fuel discharging tube

FIGURE 6. flow scheme for Helium in the core [4]

SIMULATION PROCESS

GOVERNING EQUATIONS

Thermal hydraulic calculation on HTR-10 uses ANSYS FLUENT. ANSYS FLUENT itself is one of the most popular CFD programs that are currently widely used. CFD is the study to predict fluid flow patterns, heat transfer, chemical reactions, and other phenomena by solving mathematical equations. In general, the calculation process for fluid flow is completed using the energy, momentum, and continuity equations. The equation used is the Navier Stokes equation, the equations are as follows [3]:

- Continuity Equation

$$\frac{\partial \rho}{\partial t}+\frac{\partial(\rho u)}{\partial x}+\frac{\partial(\rho v)}{\partial y}+\frac{\partial(\rho w)}{\partial z}=0 \tag{1}$$

Momentum Equation
Momentum on the Y-axis

$$\frac{\partial(\rho v)}{\partial t}+\frac{\partial(\rho uv)}{\partial x}+\frac{\partial(\rho v^2)}{\partial y}+\frac{\partial(\rho uw)}{\partial z}=-\frac{\partial p}{\partial y}+\frac{1}{Re_r}\left(\frac{\partial \tau_{xy}}{\partial x}+\frac{\partial \tau_{yy}}{\partial y}+\frac{\partial \tau_{yz}}{\partial z}\right) \tag{2}$$

Momentum on X-axis

$$\frac{\partial(\rho u)}{\partial t}+\frac{\partial(\rho u^2)}{\partial x}+\frac{\partial(\rho uv)}{\partial y}+\frac{\partial(\rho uw)}{\partial z}=-\frac{\partial p}{\partial y}+\frac{1}{Re_r}\left(\frac{\partial \tau_{xx}}{\partial x}+\frac{\partial \tau_{xy}}{\partial y}+\frac{\partial \tau_{xz}}{\partial z}\right) \tag{3}$$

- Momentum on Z-axis

$$\frac{\partial(\rho w)}{\partial t}+\frac{\partial(\rho uw)}{\partial x}+\frac{\partial(\rho vv)}{\partial y}+\frac{\partial(\rho w^2)}{\partial z}=-\frac{\partial p}{\partial y}+\frac{1}{Re_r}\left(\frac{\partial \tau_{xz}}{\partial x}+\frac{\partial \tau_{yz}}{\partial y}+\frac{\partial \tau_{zz}}{\partial z}\right) \tag{4}$$

Energy Equation

$$\frac{\partial(E_r)}{\partial t}+\frac{\partial(uE_r)}{\partial x}+\frac{\partial(vE_r)}{\partial y}+\frac{\partial(wE_r)}{\partial z}=-\frac{\partial(\rho u)}{\partial x}-\frac{\partial(\rho v)}{\partial y}-\frac{\partial(rw)}{\partial z}-\frac{1}{Re_rPr_r}\left(\frac{\partial q_x}{\partial x}+\frac{\partial q_y}{\partial y}+\frac{\partial q_z}{\partial z}\right)+\frac{1}{Re_r}\left(\frac{\partial}{\partial x}\left(u\tau_{xx}+v\tau_{xy}+w\tau_{xz}\right)+\frac{\partial}{\partial y}\left(u\tau_{xy}+v\tau_{yy}+w\tau_{yz}\right)+\frac{\partial}{\partial z}\left(u\tau_{xz}+v\tau_{yz}+w\tau_{zz}\right)\right)$$

(5)

With:

X	=	x-axis coordinates
y	=	y-axis coordinates
z	=	z-axis coordinates
u	=	velocity in x direction
v	=	velocity in y direction
w	=	velocity in z direction
t	=	time
ρ	=	Density
E_t	=	Total energy
P	=	Pressure
Q	=	*Heat Flux*
R_e	=	Reynolds number
Pr	=	Prandtl number

HTR-10 has spherical fuels that are arranged randomly on the reactor core. The random arrangement causes complexity in the flow analysis on the core of HTR-10, so an approach is needed, one of which is often used is the porous media approach. Porous media is a continuous solid phase that has a lot of space or pores in it, for example, sponges, grains of sand, cracks in hollow stones, and others. In FLUENT, porous media is modeled by adding the conditions of the momentum source conditions to the existing flow equation. The conditions for the momentum source consist of two parts, namely: viscous loss and inertia lost which can be formulated as follows [5]:

$$S_i=-\left(\sum_{j=1}^{3} D_{ij}\mu u_j+\sum_{j=1}^{3} C_{ij}\frac{1}{2}\rho[u]u_j\right) \tag{6}$$

where S_i is momentum condition at i (x, y, or z), | u | is velocity, ρ is density, and D and C are matrices. This additional momentum condition affects the pressure gradient in the porous cell and results in a pressure drop that is proportional to the velocity of fluid flow in the cell. Inhomogeneous porous media, equation (6) can be simplified into:

$$S_i = -\left(\frac{\mu}{a}u_i + C_2\frac{1}{2}\rho[u]u_i\right) \quad (7)$$

where a is the permeability and C2 is an inertial resistance factor whose value is found in the following equation

$$a = -\left(\frac{D_p^2}{150}\frac{\gamma^3}{(1-\gamma)^2}\right) \quad (8)$$

$$C_2 = -\left(\frac{3{,}5}{D_p}\frac{(1-\gamma)}{\gamma^3}\right) \quad (9)$$

where D_p is the mean diameter of the particle and ε is the void fraction in the porous cell.

HTR -10 THERMOHYDRAULIC MODELING

HTR-10 modeling using ANSYS FLUENT is carried out with a few simplifications including modeling the reactor core into two dimensions with axis-symmetry [6]. Two-dimensional modeling with axis symmetry is done because the core geometry of HTR-10 is cylindrical, so it can be approached with two dimensions where the centerline of the cylinder is the symmetry axis. Other simplifications include, in this modeling the cooling channels and also the control rod channels are not modeled, so the side reflectors are considered solid. **FIGURE 7** shows the geometry modeling for HTR-10 Core. There are several regions in two-dimensional modeling of the HTR-10 core, which include:

- Region **A** is a space above the core which has a height of 40 cm and a diameter of 90 cm. The constituent materials in region A are helium;
- Region **B** is an active core containing pebble bed fuel which has a height of 180 cm and a diameter of 90 cm. On the active core will be simulated as porous media, which contains fuel balls with a void fraction of 0.39. On this active terrace will also generate power of 10 MW;
- Region **C** is a conus containing a dummy ball made of graphite. Region C has a height of 115 cm with a diameter of 90 cm and the diameter of the output side is 25 cm;
- Region **D** is a side reflector composed of graphite material. Region D has a height of 335 cm and a thickness of the top is 120 cm, while the bottom has a thickness of 175 cm.

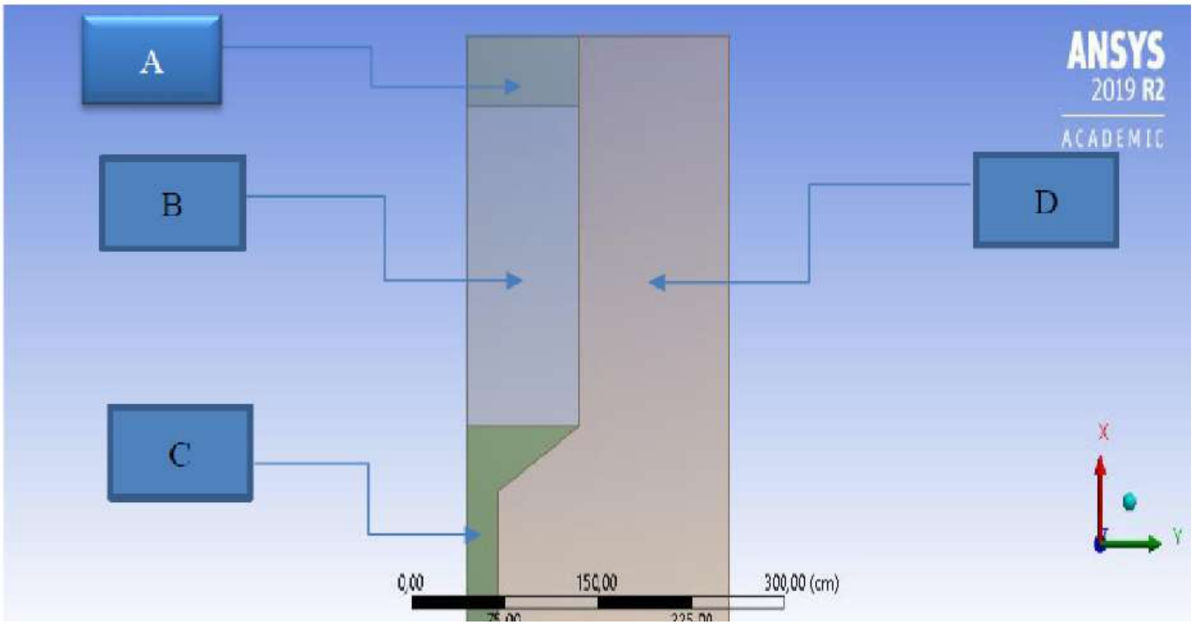


FIGURE 7. Geometry modeling for HTR-10 Core

The next step after making geometry is making mesh. The process of making mesh in ANSYS FLUENT is very important because it greatly influences the calculation results. In this modeling the mesh size used is 2 cm with the type of mesh is quadrilateral and the mesh which is at the border between the fluid and the solid is reduced to 0.2 cm. This aims to refine the calculation of heat transfer from the fluid to solid. **FIGURE 8** shows mesh for modeling on the reactor core.

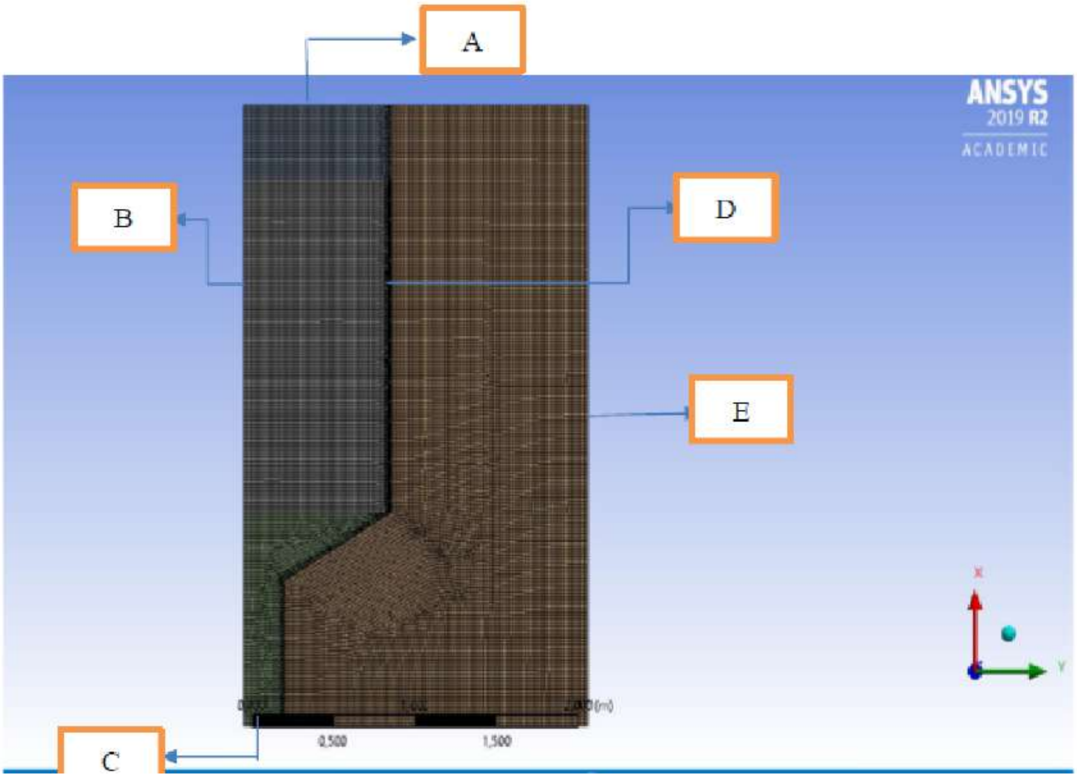


FIGURE 8. Mesh for the 2-dimensional Model of HTR-10

Boundary conditions for the reactor core model are (refer to **FIGURE 8**.):
A: Velocity inlet
B: Axis Symmetry
C: Outlet
D: Interface
E: Wall.

Helium flows from the top of the core at a speed of 3.8448 kg/s. The velocity that enters the core is 89% of the total velocity that enters the reactor vessel [7]. The cold helium will get heat from the heat generation that occurs on the active core which is modeled as porous media with helium porosity of 0.39. The helium flow then exits through the reactor core outlet and also discharged tube. **TABLE 3** shows the values for the boundary conditions for the HTR-10 model

TABLE 3. Values of boundary condition for HTR 10 Model

Boundary Condition	Value
Velocity inlet	3,8448 kg/s
Viscous resistance in porous media	261369 m^{-2}
Inertial resistance in porous media	599,89 m^{-1}
Porosity in pebble bed	0,39
Power in active core	2184303 w/m^3

RESULTS AND DISCUSSION

Thermal hydraulic simulation using ANSYS FLUENT is carried out under steady-state conditions. The power used in this simulation is at full power which is 10 MW, which for power distribution is considered uniform for all-region in the core. The active core and also the dummy ball region are represented by using porous media which uses helium gas porosity data is 0.39 and fuel porosity is 0.61.

From the results of calculations using ANSYS FLUENT data obtained include temperature distribution profiles for radial and axial directions in the core area and temperature distribution profiles in the side reflector area**. FIGURE 9** can be seen as the contour of the temperature distribution for the core area and the side reflector.

From **FIGURE 9,** it can also be seen that the temperature of helium at the time of entering the core is 250 ^{0}C while exiting the core is around 795 ^{0}C. Helium gas temperature in the dummy ball region is relatively sTABLE because in this region there is no heat generation. And the temperature of the Helium gas which is close to the reflector wall there is a significant decrease. This is because the heat from the Helium gas close to the side reflector wall will be transferred by conduction to the side reflector which contains graphite material. The distribution of helium temperature when entering the core and exiting the core at position r = 0 can be seen in **FIGURE 10**.

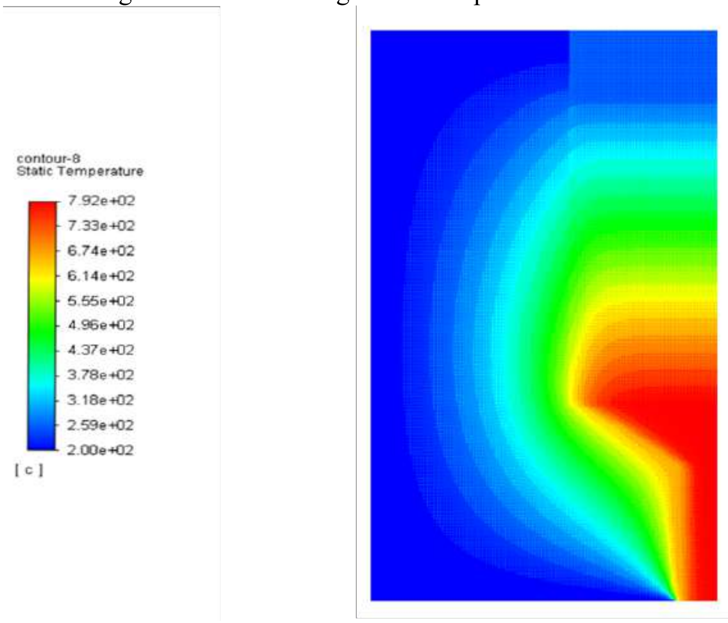


FIGURE 9. Temperature contour on HTR core

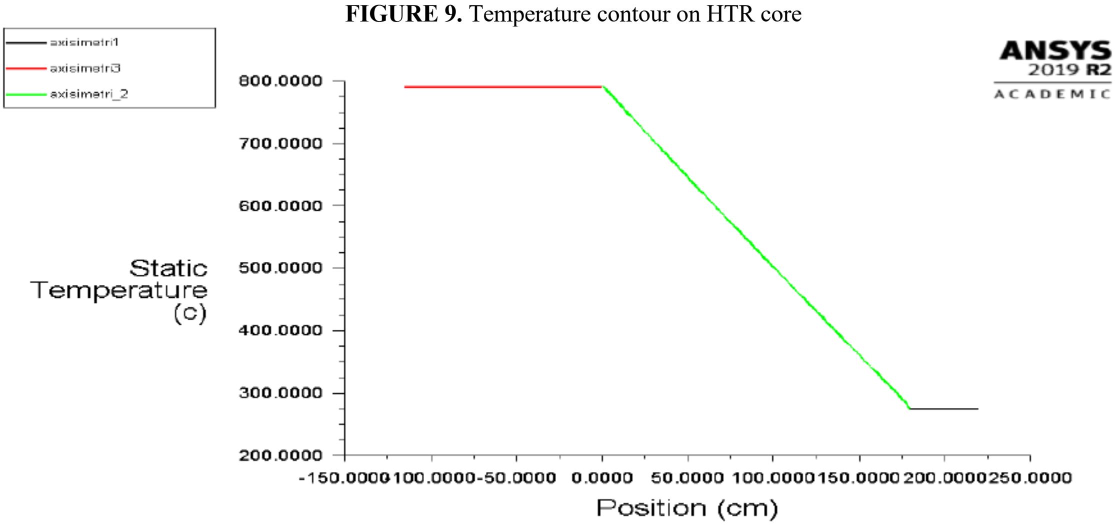


FIGURE 10. The axial direction temperature distribution at r = 0 on HTR-10 core

FIGURE 11 shows the density contour of Helium gas. The amount of density of helium gas depends on the temperature. From **FIGURE 11** it can be seen that the density of helium gas on the upper core is greater than the density of Helium gas which is on the lower core. This is because the temperature of the Helium gas on the lower core is hotter than on the upper core. For graphite, there is no change in density because graphite is solid, where the density does not change much with temperature changes.

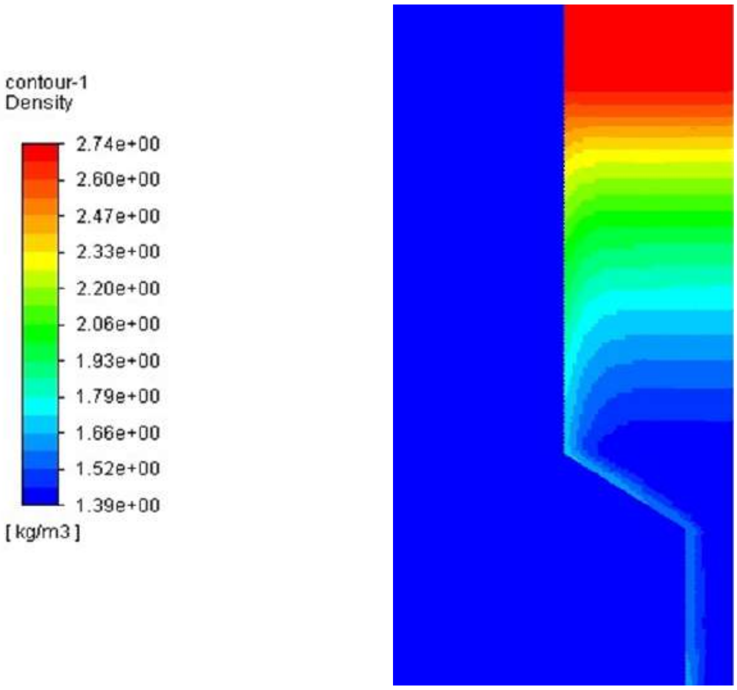


FIGURE 11. Density counter in HTR-10 core

For a comparison of FLUENT results with experimental results and with other codes, results can be seen in **TABLE 4** and **FIGURE 12** The temperatures are shown in **FIGURE 12** and **TABLE 4** is the temperature on the side reflector at a height of 80 cm (middle active core) and also 170 cm (on the output side of the active core). In **TABLE 4**, we can see the difference between the calculation results and the experimental results. The difference between the calculation and experimental results is expressed in percent using the following equation:

$$\frac{abs(code - exp)}{exp} x100\%$$

with *code* is the calculation result and *exp* is the experiment result.

There is a difference between the results of calculations using FLUENT with experimental results because there are several causes, including simplification for modeling on side reflectors, where in actual conditions there are helium gas channels and also channels for control rods which in modeling using FLUENT are not modeled. Also, there is a gap between graphite on the side reflector which can also affect heat transfer on the side reflector. The difference in results from calculations using FLUENT is not too far from differences in the results of other code calculations with experiments.

Thermal-Hydraulic calculation using ANSYS FLUENT with the porous medium approach has several disadvantages, which include the temperature displayed on the porous medium is the average temperature between the fluid and also the solid that is in the porous medium. So that calculations with the porous media approach cannot calculate the maximum temperature of pebble fuel in the core

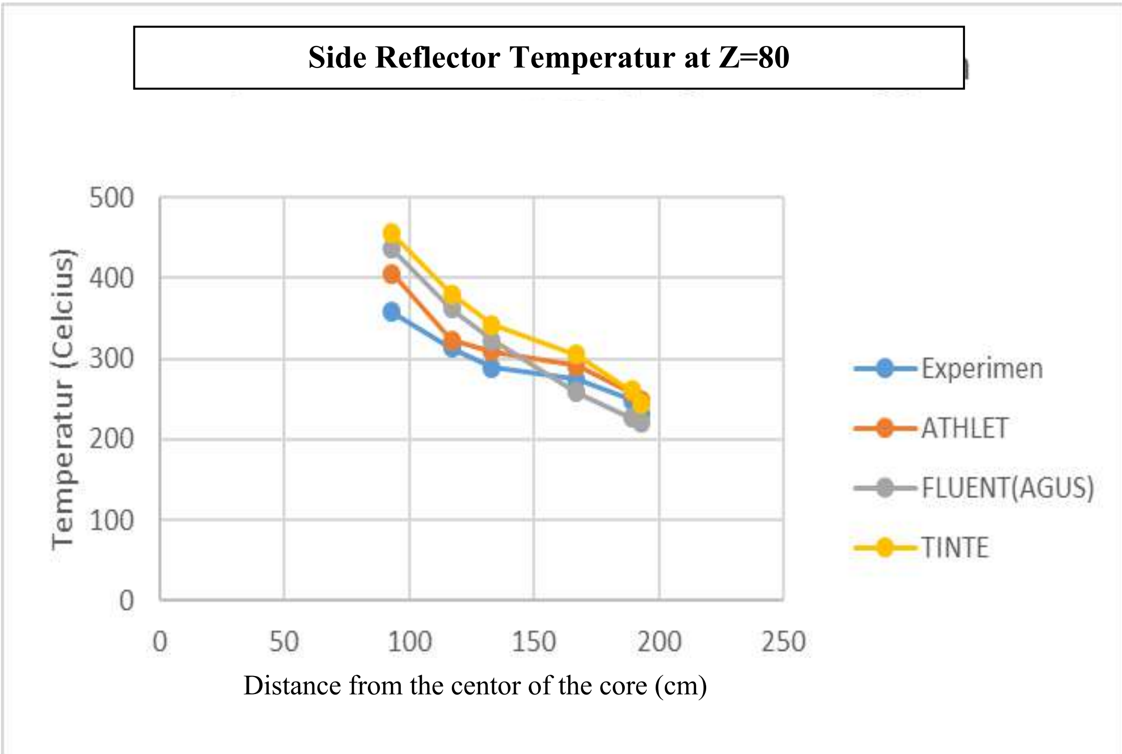
CONCLUSION

From the results of this study, several conclusions can be summarized, including that the hydraulic thermal analysis for the HTR-10 can be carried out using the ANSYS FLUENT program with a porous media approach. From the results of calculations using ANSYS FLUENT the following results were obtained:

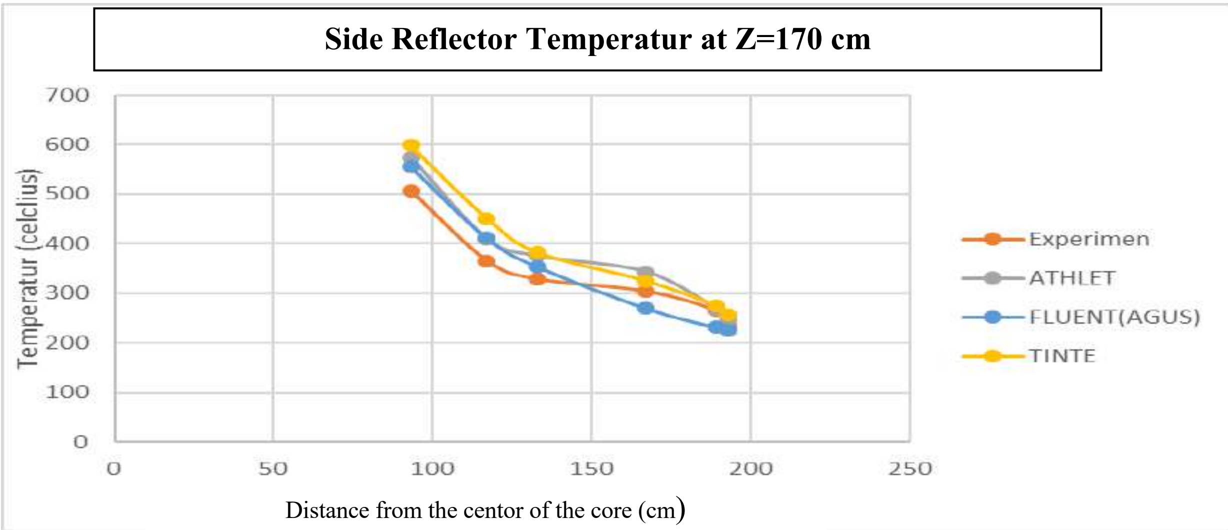
- The temperature of helium gas at the exit core is 795 0C, which is the average temperature between helium gas and pebble fuel on the terrace.
- For the measurement of temperature in the side reflector composed of graphite, the results of calculations with ANSYS FLUENT at a height of 80 cm have the least difference with the experimental results is 4% (at a radius of 193 cm) and the biggest difference is of 22% (at radius 93 cm). And for the height of 170 cm, the least difference with experiments is 4,3% (at a radius of 93 cm) and the greatest difference is at 12,76% (at a radius of 189 cm).

TABLE 4. Comparison experiment and code result

Comparation Experiment And Code												
No	R (cm)	Z (cm)	Exp. (oC)	Calc. (oC)	Diff(oC)	Dev (oC)	Calc. (oC)	Diff(oC)	Dev (oC)	Calc. (oC)	Diff(oC)	Dev (oC)
				ATHLET			FLUENT(AGUS)			TINTE		
Side reflector			Experimen									
1	193	80	231,3	248,7545	17,4545	7,55%	221,37	9,93	4%	243,9	12,6	5%
2	189	80	249,3	258,0121	8,7121	3,49%	226,69	22,61	9%	260,2	10,9	4%
3	167	80	274,3	291,8324	17,5324	6,39%	258,94	15,36	6%	305,4	31,1	11%
4	133	80	289,1	309,264	20,164	6,97%	322,91	33,81	12%	342	52,9	18%
5	117	80	313,3	323,1624	9,86235	3,15%	361,83	48,53	15%	380,4	67,1	21%
6	93	80	357,7	405,8235	48,12345	13,45%	436,26	78,56	22%	456	98,3	27%
7	193	170	234,9	248,7031	13,8031	5,88%	224,79	10,11	4,30%	257,1	22,2	9,45%
8	189	170	264,8	268,9511	4,1511	1,57%	231,02	33,78	12,76%	274,4	9,6	3,63%
9	167	170	303,9	343,3283	39,4283	12,97%	269,61	34,29	11,28%	324,3	20,4	6,71%
10	133	170	328,5	375,0942	46,59415	14,18%	353,14	24,64	7,50%	381,8	53,3	16,23%
11	117	170	365,3	409,4819	44,18185	12,09%	411,07	45,77	12,53%	450,9	85,6	23,43%
12	93	170	507,1	574,2257	67,1257	13,24%	555,20	48,10	9,49%	598,2	91,1	17,96%



(a)



(b)

FIGURE 12. Temperature on side reflector a) Z=80 cm, b) Z=170 cm

REFERENCES

[1] IAEA, Evaluation of High-Temperature Gas-Cooled Performance: Benchmark Analysis Related to Initial Testing of HTTR and HTR-10., Vienna: IAEA, 2003.

[2] IAEA, Evaluation of High-Temperature Gas-Cooled Reactor Performance, Benchmark Analysis Related to PBMR-400, PBMM, GT-MHRR, HTR-10, and The Astra Critical Facility, Vienna: IAEA, 2013.

[3] M. a. H. K. Versteeg, An Introduction to Computational Fluid Dynamics The Finite Volume Method, New York: John Wiley & Son Inc, 1995.

[4] P. M. D. a. J. Su, "Simulation in CFD of Pebble Bed-Advanced High-Temperature Reactor Core Using OpenFOAM," in *International Nuclear Atlantic Conference*, Belo Horizonte, Brasil, 2017.

[5] F. Inc, FLUENT 6.3 User's Guide, Centerra Resource Park: Fluent Inc, 2006.

[6] J. Vu, Modelling of Convective Heat Transfer in Porous Media, The University of Western Ontario, 2017.

[7] E. S. Sembiring, Analisis Karakteristik Aliran Fluida Primer dan Perpindahan Panas pada Teras 10 MWt High Temperature Gas Cooled Reactor -Test Module (HTR-10) pada Kondisi Ajek dan Transien Menggunakan FLUENT 6.3 Model 3 Dimensi, Yogyakarta: Departemen Teknik Nuklir dan Teknik Fisika Fakultas Tekni Universitas Gadjah Mada, 2018.

Simulation and Computation of the Formation Reactor's Poison: Xe-135 and Sm-149 in Thorium Molten Salt Reactor-500 with MCNP6 Code

Lee Youri Mikhaelia[1], Riyatun[1], Suharyana[1], and Azizul Khakim[2]

[1]*Department of Physics, Faculty of Mathematics and Natural Science, Universitas Sebelas Maret, Surakarta, Indonesia*
[2]*Nuclear Energy Regulatory Agency of Indonesia (BAPETEN)*

Corresponding author: suharyana61@mipa.uns.ac.id

Abstract. Simulation and computation of Thorium Molten Salt Reactor 500 (TMSR-500) have been carried out using the MCNP6 code, referring to the design of MSR-ThorCon by Martingale Inc. USA. Burn up on MSR is run on a high-performance computer. MSR uses fuel as well as the reactor coolant of a liquid mixture of BeF2-NaF-ThF4-UF4. An important information from nuclear reactor operations is the rate of formation of poison nuclides for nuclear reactors, namely Xe-135 and Sm-149, because it has a very high cross section of neutron absorption, respectively 2.65×10^6 barn and 4.014×10^4 barn. The formation rate of Xe-135 reaches a constant value of 1.65×10^{15} a / cm^3-hour, at the reactor operating time is 50 hours and the k_{eff} is 0.98668. The Sm-149 formation rate reaches a constant value of 1.03×10^{17} a / cm^3-hour at 600 hours operating time with a k_{eff} of 0.96501. When the reactor is shutdown, a Xe-135 peak will be called the xenon dead time, which at this time the reactor should not be turned on because it will cause a neutron buildup. The reactor may be restarted after the Xe-135 concentration equal to its equilibrium, that is after at least 30 hours.

Keywords: Thorium Molten Salt Reactor 500 (TMSR 500), MCNP6, Xe-135, Sm-149.

INTRODUCTION

Thorium Molten Reactor – 500 (TMSR-500) is a reactor designed for the Indonesian market based on MSR-ThroCon by Martingale Inc., USA. Unlike the conventional reactors that use solid fuel, TMSR-500 uses liquid as it fuel. The liquid which is a mixture of liquid salt and nuclear fuel is not only used as a fuel but also used as the reactor's primary coolant [1].

The main reaction that occurs inside the reactors is a controlled chain fission reaction. A fission reaction is a process where a heavy nuclide will split into two lighter nuclides [2]. From the fission reaction, many nuclides called reactor poison accumulated in the core. Reactor poison is a neutron absorber because it has a large cross-section of neutron absorption, so it can cause negative reactivity and decrease in reactor criticality. This can result in a reduction of the fission chain reaction and can cause the reactor to stop operating [3]. The reactor poisons: Xe-135 and Sm-149 can pose a significant threat to the normal operation of the reactor system. So that, research about the Xe-135 and Sm-149 formation is necessary to do. This research aims to detect the Xe-135 and Sm-149 behavior inside the reactor through the simulation method using MCNP6 by modelling the one fuel log geometry to calculate the burn up and the full core geometry to calculate the reactivity.

THEORY

Xe-135 is a poison with the largest neutron absorption cross-section, which is 2.65×10^6 barn [4]. In the reactor, Xe-135 is formed in two ways, form the fission reaction (about 0.3% of the fission result is Xe-135) and from the decay of Te-135 [5]. The equation of reaction from the decay of Te-135 to become a sTABLE nuclide is as follows:

$$ {}^{135}_{52}\mathrm{Te} \underset{\substack{19.0\\ \text{second}}}{\rightarrow} {}^{135}_{53}\mathrm{I} \underset{\substack{6,6\\ \text{hours}}}{\rightarrow} {}^{135}_{54}\mathrm{Xe} \underset{\substack{9.1\\ \text{hours}}}{\rightarrow} {}^{135}_{55}\mathrm{Cs} \underset{\substack{2.3 \times 10^6\\ \text{years}}}{\rightarrow} {}^{135}_{56}\mathrm{Ba}\ (\text{stable}) \tag{1} $$

The natural removal of Xe-135 are in two ways, first is the decay of Xe-135 to Cs-135 and the second is through reaction with neutrons like the following equation [4]:

$$54135Xe + 01n \rightarrow 54136Xe + \gamma \quad (2)$$

The concentration of Xe-135 in the reactor will continue to increase until it reaches its equilibrium. The equilibrium of Xe-135 will be achieved when the rate of production of Xe-135 is equal to the rate of removal of Xe-135. The equilibrium Xe-135 concentration can be expressed in the following equation [5]:

$$N_{Xe}(equilibrium) = \frac{(\gamma_{Xe} + \gamma_I)\sum_f^{fuel} \quad \phi}{\lambda_{Xe} + \sigma_a^{Xe}\phi} \quad (3)$$

With:

N_{Xe} = Xe-135 concentration

γ_{Xe} = Xe-135 yield from fission

γ_I = I-135 yield from fission

$\sum_f^{fuel}$ = Macrosopic fission cross-section of fuel

ϕ = Thermal neutron flux

λ_{Xe} = Decay constant of Xe-135

σ_a^{Xe} = Macroscopic absorption cross-section of Xe-135

From Eq. 3, the formation rate of equilibrium Xe-135 can be written as follow:

$$\text{The formation rate of equilibrium Xe-135} = \frac{N_{Xe}}{\Delta t} \quad (4)$$

Δt is the time needed to Xe-135 reaches its equilibrium.

Besides Xe-135, another poison that needs attention is Sm-149 because it has a fairly large absorption cross-section, which is 4.014×10^4. Sm-149 is formed from fission (about 1.08% of fission) and from the decay of Nd-149. The reaction of decaying nuclides to become Sm-149 is as follows:

$$^{149}_{60}\text{Nd} \underset{\text{1.72 hours}}{\rightarrow} {}^{149}_{61}\text{Pm} \underset{\text{51.30 hours}}{\rightarrow} {}^{149}_{62}\text{Sm (stable)} \quad (5)$$

Sm-149 has different properties from Xe-135, Sm-149 is a sTABLE nuclide, so it can only be removed by absorbing neutrons into Sm-150 as in the following:

$$65149Sm + 01n \rightarrow 65150Sm + \gamma \quad (6)$$

Like Xe-135, the concentration of Sm-149 will continue to increase along the reactor operating time until it reaches its equilibrium. The value of Sm-149 concentration depends on the concentration of Pm-149. The concentration of equilibrium Sm-149 is stated in Equation 7.

$$N_{Sm}(equilibrium) = \frac{\gamma_{Pm}\sum_f^{fuel}}{\sigma_a^{Sm}} \quad (7)$$

With:

N_{Sm} = Sm-149 concentration

γ_{Pm} = Pm-149 yield from fission

$\sum_f^{fuel}$ = Macrosopic fission cross-section of fuel

σ_a^{Sm} = Macroscopic absorption cross-section of Sm-149

From Eq. 7 it can be seen that the concentration of Sm-149 in its equilibrium does not depend on the neutron flux and reactor power, but only depends on the Pm-149 yield and the absorption cross-section of the Sm-149. The rate formation of equilibrium Sm-149 is expressed by Equation 8.

The formation rate of equilibrium Sm-149 $= \frac{N_{Sm}}{\Delta t}$ (8)

Δt is the time needed to Sm-149 reaches its equilibrium.

To know the concentration of fission products during the reactor's operating period, a burn up calculation must be performed. Riyadi et al (2019) performed burn up calculations using the MCNP6 code for liquid fueled reactors, SAMOP [6]. Burn up is a standard calculation that focuses on the management and control of fuel, including measurement of used fuel, combustion process, processing and the amount of energy produced per unit weight of fuel [7].

The Monte Carlo continuous energy method in the Monte Carlo N-Particle (MCNP) code is one of the reliable methods for transporting particles such as neutrons, photons, and electrons in complex three-dimensional systems. This method is used because it can trace particles from its birth until its death [8].

METHODOLOGY

The procedure of this study is shown in **FIGURE 1.**

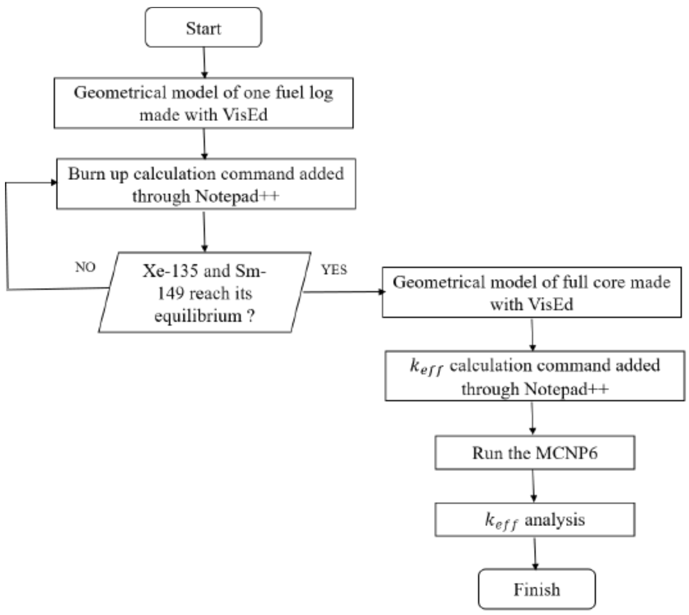


FIGURE 1. Flowchart of the research procedure

The TMSR-500 full core geometry model consists of 84 fuel logs with complex kanal. Burn up calculations with MCNP6 will take longer and larger and more complicated geometry models. To shorten the computational time without reducing the validity of the results can be used periodic boundary condition (PBC) on the fuel log geometry model [9]. The fuel log geometry model is presented in **FIGURE 2.**

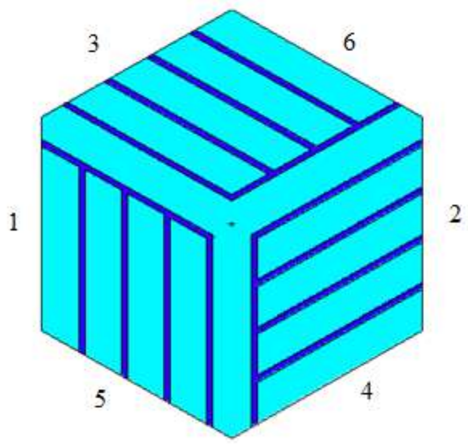


FIGURE 2. One fuel log geometrical model of TMSR-500 [9]

FIGURE 2 shows the TMSR-500 two dimensional geometry model for burn up calculations made using the Visual Editor (VisEd). Dark blue cells are mixture of BeF2-NaF-UF4-ThF4 with U-235 that enriched to 19.75%, and the light blue cells show graphite moderator.

Side 1 is periodic with side 2 and vice versa. When particles exits the log through the right side (2), it will re-enter the canal through the left side (1). The same is the case with side 3 periodic with side 4 and vice versa, and side 5 periodic with side 6 and vice versa. By applying PBC, the log that is modeled represents one fuel log that is located right in the middle of the core and is surrounded by other logs that repeat infinitely on each side. After the geometry is completed in VisEd, the burn up calculation command is added though the Notepad++, the MCNP6 is run. From the burn up results, it can be known the concentration of Xe-135 and Sm-149, which are then used to calculate the reactor criticality.

The criticality of the reactor is calculated under 2 conditions, at the beginning of the cycle (BOC) when the reactor is clean from poisons and when the reactor poison has been added. The criticality calculation is performed on the full core geometry model as presented in **FIGURE 3**. After the reactor criticality value is obtained, the reactivity calculation is done with the following equation [4]:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \tag{9}$$

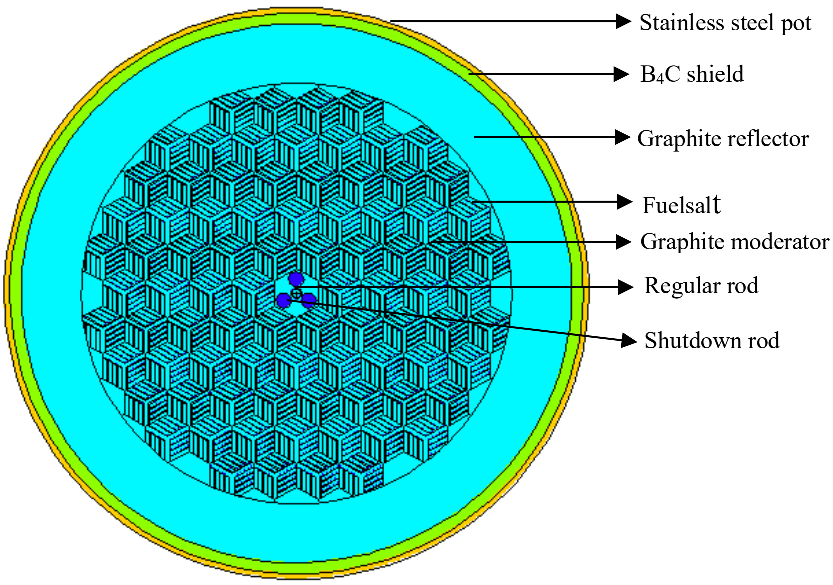


FIGURE 3. Full core geometrical model of TMSR-500 [9]

The simulated burn up time is 600 hours at full power without refueling. When it reaches 600 hours, the sudden shutdown reactor is simulated. A description of the concentrations of Xe-135 and Sm-149 is needed to explain the neutronic characteristics under these conditions. Based on the reactor poison data, ideas will emerge to overcome shutdown conditions and predict the reactor restart processes.

RESULTS AND DISCUSSION

Burn up calculations in MCNP were performed using KCODE with 10^5 simulated particles within 250 cycles and 35 of them being skipped. The only burn up material is fuel. The calculation is done using a high-performance computer and requires 7 days of running.

The k_{eff} value of the one fuel log model is (1.01643 ± 0.00009), and for the full core model, the k_{eff} is (1.01885 ± 0.00008). The difference in the k_{eff} of both models is not too significant, which is 0.238% so that it can be said that the geometry model of the one fuel log correctly represents one log that is located right in the middle of the core. The k_{eff} value in the both geometry model of this study is very close to k_{eff} claim value of TMSR-500 [9] which is 1.00092, with a difference of less than 2%.

FIGURE 4 shows the concentration of Xe-135 against time. At t=0, the concentration of Xe-135 is also 0 which indicates that the reactor is clean and free of reactor poison. At intervals of 0 to 10 hours, the concentration Xe-135 builds up linearly from 0 to 4.74 x 10^{16} $^{a}/cm^3$. This build up is caused by the contribution of U-235 fission and Te-135 decay (Te-135 to Xe-135 requires around 6.6. hours). At intervals of 10-30 hours, the concentration of Xe-135 shows that the accumulation of Xe-135 has begun to be equal to its removal by neutron absorption to formed Xe-136 and beta decay to formed Cs-135. At 30-50 hours, the concentration of Xe-135 reaches its equilibrium.

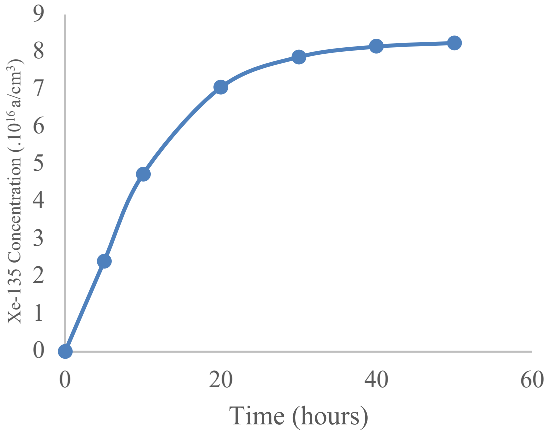


FIGURE 4. Xe-135 concentration at reactor startup

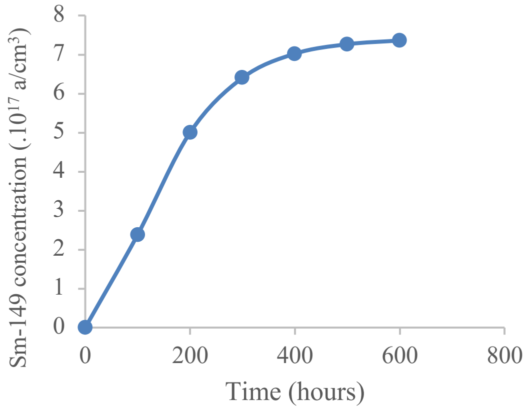


FIGURE 5. Sm-149 concentration at reactor startup

FIGURE 5 shows the Sm-149 concentration against time. At t=0, the Sm-149 concentration also 0, it means that at the beginning of the operation, there's no reactor poison in the core. From 0 to 400 hours of operation, Sm-149 increased from 0 to 7.02 x 10^{17} a/cm^3. From the graph of Sm-149 profile, it can be seen that Sm-149 reaches its equilibrium after 400 hours. Sm-149 takes longer to reach its equilibrium than Xe-135, because aside from the fission product, Sm-149 is also produced by Nd-149 decay that needs about 54 hours to become Sm-149.

FIGURE 6 shows the comparison of Xe-135 and Sm-149 against time at the first 30 days of reactor operation with constant power 250 MW.

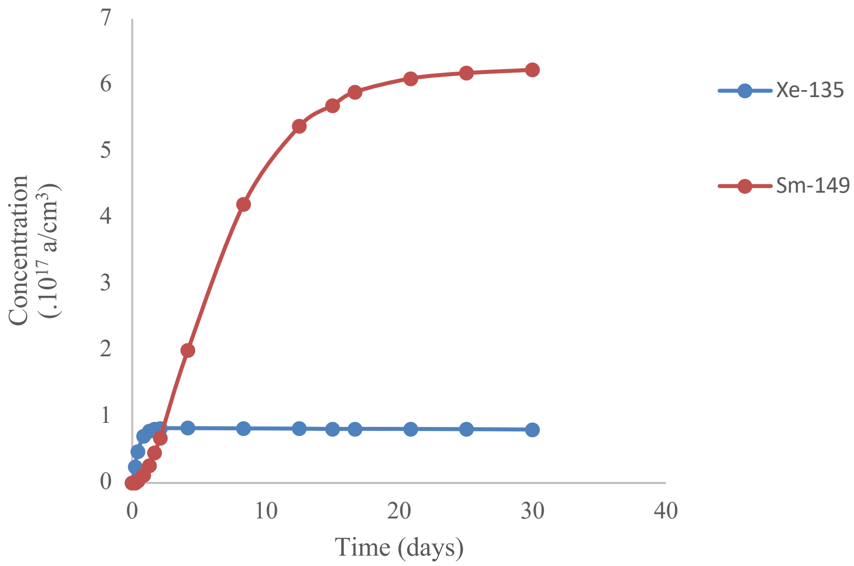


FIGURE 6. Comparison of Xe-135 and Sm-149 at the first 30 days of reactor operation

From **FIGURE 6**, it can be seen that when Xe-135 reaches its equilibrium on the second day of reactor operation, Sm-149 still increasing. Sm-149 reaches its equilibrium after about 20 days of reactor operation. Without the changes in power level, the fission rate of the reactor almost constant for all cycles. As a result, throughout the reactor operation, the concentration of Xe-135 is 1.65x10^{15} a / cm^3-hour and Sm-149 is 1.03x10^{17} a / cm^3-hour. This number is almost the same as the research of Intokiyah and Subhki (2019) that study about reactor poison using GUI [10]. Xenon is a gas poison, so it should be extracted from the core. On TMSR-500, Xe-135 is extracted using spray bubbling methods every 25 minutes with flowing 0.45 grams/hour of helium at the Primary Loop Pump [9].

The next simulation is a condition when the reactor shutdown suddenly (without minding the cause). **FIGURE 7** and **8** shows the concentration of Xe-135 and Sm-149 when reactor shutdown until restarted.

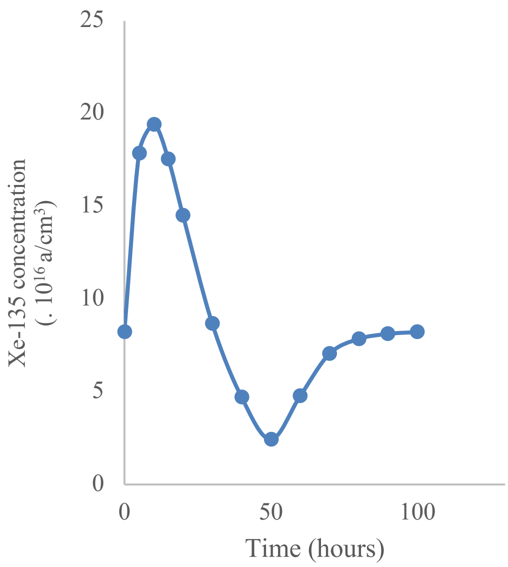


FIGURE 7. Xe-135 concentrations at reactor shutdown and restart

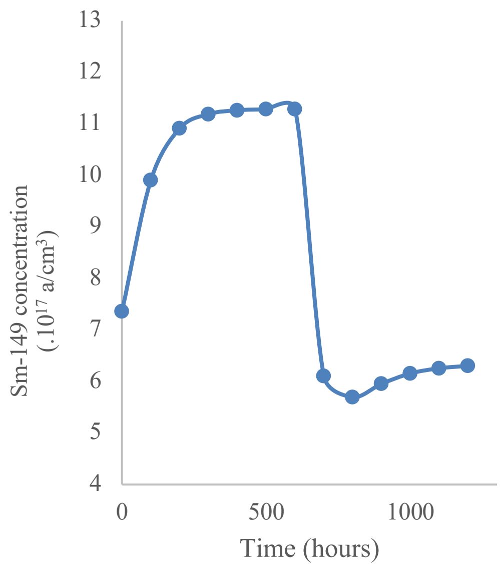


FIGURE 8. Sm-149 concentrations at reactor shutdown and restart

When the reactor is shutdown, the power will drop to 0 MW. This causes the neutron flux in the reactor to drop drastically to zero. Because the neutron flux is zero, there's no Xe-135 formed from fission. However, Xe-135 is still formed from the decay of I-135. **FIGURE 7** shows 10 hours after the shutdown, xenon peak is formed with a concentration of 1.94×10^{17} a/cm^3. The xenon peak is formed because the decay rate of I-135 is faster than the decay rate of Xe-135, and Xe-135 cannot absorb neutrons due to the unavailability of free neutrons. However, the xenon peak does not last long because after a while the amount of Xe-135 formed from the decay of I-135 will decrease and become smaller than the rate of Xe-135 decay so that its concentration will decrease. The height of the xenon peak depends on the magnitude of the neutron flux before the reactor is turned off. The higher the neutron flux value, the higher the peaking.

When the reactor is shutdown, the Sm-149 concentration will not form a peak like Xe-135. The Sm-149 concentration will slowly increase until it reaches a maximum value of 1.13×10^{18} a/cm^3 after the reactor has been turned off for 600 hours. When the reactor power is raised again to 250 MW, the Sm-149 will again absorb neutrons and return to its equilibrium. From **FIGURE 8** we can also know that the reactor will never clean from Sm-149 unless its fuel is renewed because Sm-149 is stable.

The reactivity of Xe-135 and Sm-149 was obtained from the calculation of the difference in reactivity of the reactor when its fuel was clean and when the concentration of Xe-135 or Sm-149 was added. The reactivity values of the two poisons are presented in **TABLE 1**.

TABLE 1. Xe-135 and Sm-149 reactivity

Nuclide		k_{eff}	Reactivity ($\% \frac{\Delta k}{k}$)
Fresh fuel		1.01885	1.850
Xe-135	Equilibrium	0.98454	-3.423
	Peak	0.94221	-7.985
Sm-149	Equilibrium	1.01285	-0.584
	Peak	1.00815	-1.044

Although in FIGURE 6 the concentration of equilibrium Sm-149 is much higher than Xe-135, **TABLE 1** shows that the reactivity value of equilibrium Xe-135 is greater than Sm-149. This is because the Xe-135 absorption cross-section that is 2.65×10^6 barn is larger than the Sm-149 absorption cross-section that is 4.014×10^4 [4], so Xe-135 will absorb more neutrons than Sm-149.

At the peak of Xe-135, the negative reactivity was 57.13% more than the equilibrium Xe-135 reactivity. Positive reactivity is needed which is proportional to the negative reactivity so that the reactor can be operating again. Positive reactivity can be added by lifting the control rod from the core or by adding fissile material. The period time when the reactor cannot return to operation as a result of the Xe-135 influence is called xenon dead time. During the xenon dead time period, positive reactivity should not be added because there can be a large accumulation of neutrons. This is very dangerous because the concentration of Xe-135 continues to decrease until at a certain point the reactor can suddenly start and explode because it cannot handle the accumulation of neutrons that occurs. From **FIGURE 7** it can be seen that the concentration of Xe-135 almost equal to its equilibrium after 30 hours of shutdown. Thus the save minimal time to restarting the reactor is 30 hours.

CONCLUSION

The main reactor poison, Xe-135, reaches equilibrium after 50 hours of reactor operation with a value of $1.65x10^{15}$ a / cm^3-hour with a reactivity contribution of -3.423 $\%\frac{\Delta k}{k}$. The second poison, Sm-149 reached the equilibrium after 600 hours of reactor operation with a value of $1.03x10^{17}$ a / cm^3-hour with a reactivity contribution of -0.584 $\%\frac{\Delta k}{k}$. After reaching equilibrium, the concentration of Xe-135 and Sm-149 will tend to be constant if there is no change in reactor power. When the reactor is shutdown, a Xe-135 peak will be called the xenon dead time, which at this time the reactor should not be turned on because it will cause a neutron buildup. The reactor may be restarted after the Xe-135 concentration equal to its equilibrium, that is after at least 30 hours.

REFERENCES

[1] J.D. Thomas, *Molten Salt Reactor and Thorium Energy 1st ed* (Woodhead Publishing, Cambridge, MA, USA, 2017).

[2] E. Fermi, Nucl. Disitegr. Electr. Engin. **59(2)** 57-58 (1940).

[3] M.A. Jayeola et al. Bilg. Intl. Jrnl. Sci. Tech. Res. **2(1)** 17-30 (2018).

[4] DOE Fundamentals Handbook, *Nuclear Physics and Reactor Theory* vol 2 (Department of Energy, Washington DC, 1993).

[5] J.R. Lamarsh and J.A. Baratta, *Introduction to Nuclear Engineering, Third Edition* (Prentice-Hall, USA, 2001).

[6] R. Riyadi, Suharyana, F. Anwar, Riyatun, and Soeparmi, Analisis Aktivitas Nuklida Hasil Burn Up pada Fasilitas Subcritical Assembly For Mo99 Production (SAMOP) PSTA-BATAN Menggunakan Software ORIGEN2 (2019).

[7] W. M. Stacey, *Nuclear Reactor Physics* (John Wiley & Son, New York, 2007).

[8] Y. J. Mostaar, E. Caluguru, and E. Torre, Jrnl. Rad. Res **3** 143-149 (2003).

[9] J. Devanney, *ThorCon™-The Do-able Molten Salt Reactor* (ThorCon USA Inc, USA, 2019).

[10] D. Intokiyah and M. N. Subkhi, “Simulasi Peracunan Xenon dan Samarium Menggunakan GUI” in Prosiding Seminar Nasional Fisika (SINAFI) (2018).

Effect of Inner Shell Diameter and Fuel Salt Concentration on Criticality and Conversion Ratio of CAMOLYP Reactor

Ghulam Abrar[1, a)] and Syarip[2, b)]

[1]Department of Nuclear Engineering and Engineering Physics, Jl. Grafika No. 2, Kampus UGM, Yogyakarta 55281

[2]Center for Accelerator Science and Technology, National Nuclear Energy Agency (BATAN), Jl. Babarsari, Yogyakarta 55281

[a)] gh.abrar@gmail.com
[b)] syarip@batan.go.id

Abstract. PSTA (Center for Accelerator Science and Technology) BATAN (National Nuclear Energy Agency) Yogyakarta is currently developing a concept critical low power reactor or Critical Assembly for ^{99}Mo Isotope Production (CAMOLYP). The concept of CAMOLYP reactor core is based on the ^{235}U or ^{233}U fission reaction process, so at the beginning of the operation, these elements are needed. The next cycle of operation no longer requires the ^{235}U or ^{233}U enrichment process because it is expected that ^{233}U will be fulfilled through the breeder process of thorium (^{232}Th). For this purpose, research on criticality and conversion ratio (CR) analysis is needed in the CAMOLYP design. If CAMOLYP is capable of being critical with a CR value of more than one, then CAMOLYP can be categorized as a thermal breeder reactor which means that CAMOLYP can utilize thorium as a substitute for ^{235}U as its fuel. The aims of this study to determine the effect of changes in inner shell diameter and concentration of fuel solution on multiplication factors and conversion ratios in two CAMOLYP core reactor models, i.e. Design-A and Design-B. Design-A is a proposed CAMOLYP reactor core design and Design-B is a model that uses the core and blanket concepts. The method used is a neutronic calculation by using MCNPX 2,7 computer code. The current CAMOLYP design can't be able to become a thermal breeder reactor. The CAMOLYP reactor core with Design-A can achieve its criticality condition with CR of 0,203644. While CAMOLYP with Design-B also can achieve its criticality with the highest CR of 0,1309661. Based on the analysis results it can be concluded that to increase k_{eff} value, it can be done by enlarging the inner shell diameter for Design-A and enlarging the core diameter for Design-B. Another way to increase k_{eff} value is by increase fissile content by adding ^{235}U content in TRIGA 104 fuel or ^{235}U concentration in thorium-uranyl nitrate solution. The value of CR can be increased in a way increasing the concentration of thorium in fuel solution. Increasing blanket thickness does not affect CR

Keywords: CAMOLYP, Thermal Breeder Reactor, Conversion Ratio, Thorium, Mo-99.

INTRODUCTION

Various reports have indicated that there is a global shortage of ^{99}Mo production. The availability of these radioisotopes is also increasingly under threat due to the shutdown of several ^{99}Mo production research reactors and processing facilities [1]. The ^{99}Mo production is normally carried out by the target irradiation technique by a high-intensity neutron source originating from a research reactor. The target is in the form of high enriched uranium, which is greater than 90% ^{235}U so that the target material is highly enriched uranium and a nuclear reactor with a high neutron flux as well. The target material which has been irradiated with neutrons is then dissolved and then ^{99}Mo fission products are extracted from the solution [2-4]. The use of high-enriched targets has been criticized for the risks of abuse associated with nuclear security and proliferation. Therefore, the IAEA encourages the production of medical isotopes using low enriched targets. However, the use of low enriched targets is not economically attractive because will reduce the amount of ^{99}Mo that can be produced. From these deficiencies, came the idea of developing a reactor with fuel in the form of a solution or Aqueous Homogeneous Reactor (AHR) [5].

PSTA (Center for Accelerator Science and Technology) BATAN (National Nuclear Energy Agency) Yogyakarta took part in the development of AHR for radioisotope production and currently developing a concept critical low power reactor or Critical Assembly for ^{99}Mo Isotope Production (CAMOLYP). The concept of CAMOLYP reactor core is based on the ^{235}U or ^{233}U fission reaction process, so at the beginning of the operation, these elements are needed. The next cycle of operation no longer requires the ^{235}U or ^{233}U enrichment process

ISSN: 1412-3258 © 2020 BAPETEN – FMIPA UI

because it is expected that ^{233}U will be fulfilled through the breeder process of thorium (^{232}Th). Based on the CAMOLYP concept, it is expected that the dependence on ^{235}U material which has been very limited and strict in use, can be fulfilled with ^{233}U which has almost the same physical characteristics [6-8]. For this purpose, research on criticality and conversion ratio (CR) analysis is needed in the CAMOLYP design. If CAMOLYP is capable of being critical with a CR value of more than one, then CAMOLYP can be categorized as a thermal breeder reactor which means that CAMOLYP can utilize thorium as a substitute for ^{235}U as its fuel [9].

Thorium is claimed more environmentally friendly and the risk of possible waste being used as nuclear weapons is very low [2,10]. As a breeder that produces uranium which can be used as nuclear reactor fuel, thorium can also indirectly produce radioisotope molybdenum (^{99}Mo) which is one of the most strategic materials in medical activities in the field of nuclear medicine. The ^{99}Mo isotope is the ^{99m}Tc radioisotope generator which is the most widely used radioisotope for diagnostics in the field of nuclear medicine [3,10].

In a fast breeder reactor, CR values are relatively high with a value of 1,3-1,5. In the thermal breeder reactor, the margin for breeding is extremely small so that the physics of the thermal breeder reactor is dominated by the neutron economy. The neutron economy is improved by reducing neutron loss. The reduction of neutron loss is done by surrounding the fissile material on the seed region with fertile material on the blanket [11].

The light water breeder reactor using the ^{232}Th-^{233}U fuel cycle has been successfully demonstrated by the Shippingport reactor. The report entitled "The Shippingport Pressurized Water Reactor and Light Water Breeder Reactor, WAPD-T-3007" was written by J.C. Clayton in 1993 [12] explained that after completion of operations, an analysis of the fuel rods in the Shippingport reactor core was carried out either through a destructive test or a non-destructive test. The analysis results show that the reactor core contains 1.39% more fissile material than the initial operation of the reactor after operating for 5 years.

The Shippingport light water reactor uses ^{233}U fuel on the seed and ^{232}Th on the blanket. The fuel module consists of a hexagonal seed module that is surrounded by a blanket module including at the top and bottom. Some of the neutrons produced by the seed are used to carry out chain reactions while the neutrons that come out of the seed will be reflected by the reflector or absorbed by the fertile material [13].

As reported in Ref [14], "Neutronics Analysis of Aqueous Homogeneous Reactor (AHR) Subcritical Breeder Reactor to Produce Mo-99 with Thorium Sulfate Fuel (ThO_2 (SO_4))" A. Sagita has succeeded in designing a subcritical system with CR value above 1. The AHR design has a spherical shape with the blanket surrounding the core. A. Sagita concluded that the optimal design was to use a fuel solution with uranyl sulfate and thorium nitrate mole fraction of 22.5% with a size ratio of cores and blankets are 1: 2. The CR value obtained was 1,10349.

The concept of the CAMOLYP reactor core was developed from the previous SAMOP (Sub-critical Assembly for ^{99}Mo Isotope Production) concept [4,15,16]. From the previous research, it was shown that using only thorium-uranyl nitrate with a graphite reflector of 20 cm thickness in a spherical shaped reactor would need 179.5 L of thorium-uranyl nitrate to achieve criticality [8,15,16]. This was relatively large compared to SAMOP [4] or CAMOLYP [7,8], with a volume of 20 L and 56.7 L respectively. Because using only thorium-uranyl nitrate is considered to be unsuiTABLE as it needs a large amount of fuel, a new study is being conducted by using uranyl-nitrate as its fuel and thorium-uranyl nitrate as its blanket.

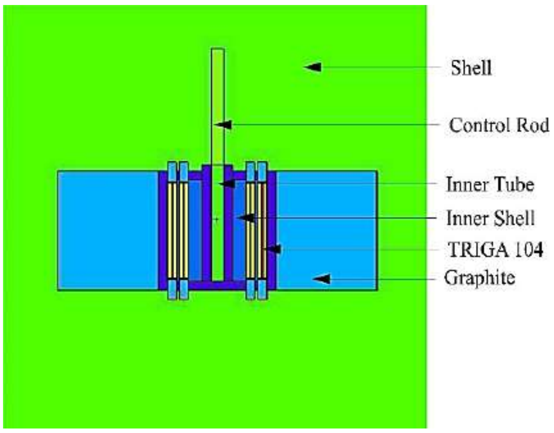
The CAMOLYP core design has not been proven to be a thermal breeder reactor. To obtain the expected CAMOLYP reactor core design, a lot of initial research is needed regarding the design performance and safety. Various CAMOLYP core designs needed to be analyzed so they have a good neutron economy to become a thermal breeder reactor. The aims of this study to determine the effect of changes in inner shell diameter and concentration of fuel solution on multiplication factors and conversion ratios in two CAMOLYP core reactor models. The first model is a proposed CAMOLYP reactor core design and the second model is a model that uses the core and blanket concepts as has been implemented in the literature review above. The results of this study are expected to provide an overview of the CAMOLYP design that is capable of breeding as well as preliminary research for further CAMOLYP breeder studies.

DESCRIPTION OF CAMOLYP REACTOR CONCEPT

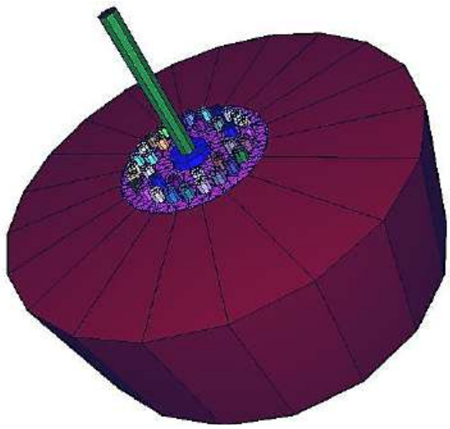
CAMOLYP is a modular reactor that can be installed module by a module having shell and tube type. The tube part consists of an inner tube and an inner shell. The tube is made of SS-316 alloy. Above the inner tube is a control rod. The inner shell contains TRIGA fuel type-104 and thorium-uranium nitrate or sulfate solution. The shell part of the CAMOLYP reactor is cooling water. TRIGA fuel type 104 and thorium-uranium nitrate or sulfate solution have a function as fuel and also as a target for ^{99}Mo production. The proposed CAMOLYP reactor core design (Design-A) specification is shown in **TABLE 1**, and a schematic diagram is shown in **FIGURE 1**.

TABLE 1. CAMOLYP rector core specification (Design-A).

Parameter	Value / Materials	Unit
Fuel 1	Thorium-Uranyl Nitrate (60% Th : 40% U)	
Fuel 2	TRIGA 104	
Fuel 1 and Fuel 2 Enrichment	19.75	%
Inner Tube Diameter	5	cm
Inner Tube Height	46	cm
Inner Shell Diameter	40-70	cm
Inner Shell Height	40	cm
Number of Fuel Rods	12-36	
Reflector Thickness (Graphite)	40	cm
Coolant	Demineralized Water	
Tube Framework (SS-316) Thickness	3.6	cm
Average Neutron Flux	10^9-10^{10}	neutron/cm^2 s
Reactor Power	1	kW



(a)



(b)

FIGURE 1. CAMOLYP reactor core of Design-A a schematic diagram (a), and a 3D model (b).

The second design is Design-B, it is developed from Design-A. In this design, the inner shell will be divided into two parts namely core and blanket. The core will be filled with Uranyl Nitrate solution with 19.75% enrichment and the blanket will be filled with Thorium Nitrate solution. The specification of Design-B is described in **TABLE 2**, and its schematic diagram is shown in **FIGURE 2**.

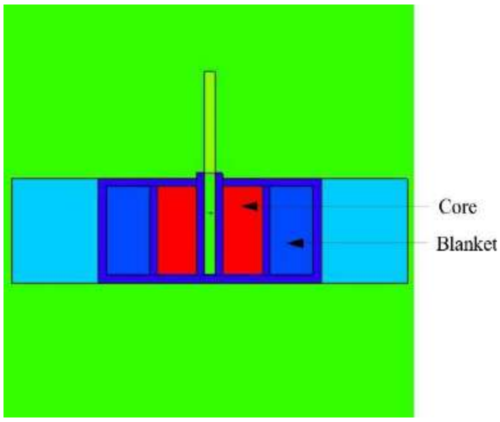


FIGURE 2. Design-B CAMOLYP reactor core arrangement.

TABLE 2. CAMOLYP rector core specification (Design-B)

Parameter	Value / Material	Unit
Core	Uranyl Nitrate	
In-Core Fuel Enrichment	19.75	%
Blanket	Thorium Nitrate	
Inner Tube Diameter	5	cm
Inner Tube Height	46	cm
Core Diameter (Inner Shell 1)	40-50	cm
Blanket Thickness (Inner Shell 2)	20-25	cm
Inner Shell Height	40	cm
Reflector Thickness (Graphite)	40	cm
Reactor Coolant	Demineralized Water	
Tube Framework Thickness (SS-316)	3.6	cm
Average Neutron Flux	10^9-10^{10}	neutron/cm^{2}s
Reactor Power	1	kW

METHODS

The method used in this study is creating the CAMOLYP reactor core model according to design specification on the MCNPX 2,7 input to evaluate the neutronic behavior such as criticality and conversion ratio. MCNPX (Monte Carlo N-Particle eXtended) is a computer code that simulates various types of particle interaction (radiation transport) over a wide range of energy. MCNP was developed by Los Alamos National Laboratory since 1957[17].

Multiplication factor (k_{eff}) in MCNP is defined as [18]

$$k_{eff} = \frac{N \int_V \int_0^\infty \int_E \int_\Omega \nu\sigma_f \phi dV dt dE d\Omega}{\int_V \int_0^\infty \int_E \int_\Omega \nabla \cdot J dV dt dE d\Omega + N \int_V \int_0^\infty \int_E \int_\Omega (\sigma_c + \sigma_f + \sigma_m)\phi dV dt dE d\Omega} \tag{1}$$

where V, t, E, and Ω are variable for volume, time, energy, and neutron direction. N is atomic density and σ is the microscopic cross section. The numerator in Eq. 1 represents neutron production at fission and the denominator represents neutron losses which are the summation of the neutron leaks, capture rate (n,gamma), fission rate (n,fission), and multiplication rate (n,xn).

The control parameters in the MCNPX calculation used in this study are source size per cycle is 10000, the initial guess for k_{inf} is 1, the number of inactive cycles is 10, and the total cycles is 220.

CR is defined as the ratio between fissile material produced and fissile material lost [19]. At the beginning of the CAMOLYP cycle, there are ^{235}U fissile material and ^{232}Th and ^{238}U fertile material. CR values are evaluated in the following equation [19]:

$$CR = \frac{neutron\ absorption\ reaction\ rate\ in\ fertile\ material}{neutron\ fission\ and\ absorption\ reaction\ rate\ in\ fissile\ material} \tag{2}$$

$$reaction\ rate = \phi\ .\ \sigma\ .\ N\ .\ V \tag{3}$$

where ϕ is neutron flux. CR calculation is done by input burnup function in MCNP so that MCNP will provide collision rate information. Collision rate on specific nuclide will be used to calculate CR, so Eq. 2 become

$$CR = \frac{collision\ rate\ (n,gamma) 90232Th + collision\ rate\ (n,gamma)\ 92238U}{collision\ rate\ (n,gamma)\ 92235U + collision\ rate\ (n,fission) 92235U} \tag{4}$$

MCNP simulates all neutrons histories on specified geometry from birth by fission to death by escape, parasitic capture, or absorption. From this simulation, MCNP obtains a collision rate for all nuclides. When simulate neutrons, MCNP using an interaction TABLE for each isotope to determine neutrons motion and behavior [20]. In this study, the ENDF/B-VII.1 data library was used.

The criticality calculation of the reactor core model is performed first. If the core can't achieve criticality condition, then the analysis is carried out to determine the next design variation needed to obtain critical condition. After critical condition was achieved, a CR calculation is performed. The result from CR calculations is also analyzed to consider the next design variation to obtain CR values as high as possible.

In this study, two CAMOLYP core reactor models will be analyzed. The first model is Design-A, which is the proposed CAMOLYP design as described in **TABLE 1**. In Design-A, the variation of inner shell diameter and number of fuel rods will be carried out. Furthermore, the variation of uranium content in TRIGA fuel and concentration of thorium-uranyl nitrate is also be performed. The second model is Design-B. In Design-B, the variation of core diameters and blanket thickness will be carried out as well as the variation of uranyl nitrate and thorium nitrate concentrations. For each variation, the effect on changes in k_{eff} and CR will be investigated.

RESULT AND DISCUSSION

Design-A Analysis

Before calculating CR values, it is necessary to ascertain in advance whether the CAMOLYP reactor has reached a critical condition or not. The first study is carried out by varying inner shell diameter and number of fuel rods. In this study, TRIGA type 104 fuel with 8.5% uranium content and thorium-uranyl nitrate concentration 200 g Th-U/L are used and kept at constant [21,22]. The criticality calculation results for various reactor core configurations are shown in **FIGURE 3**.

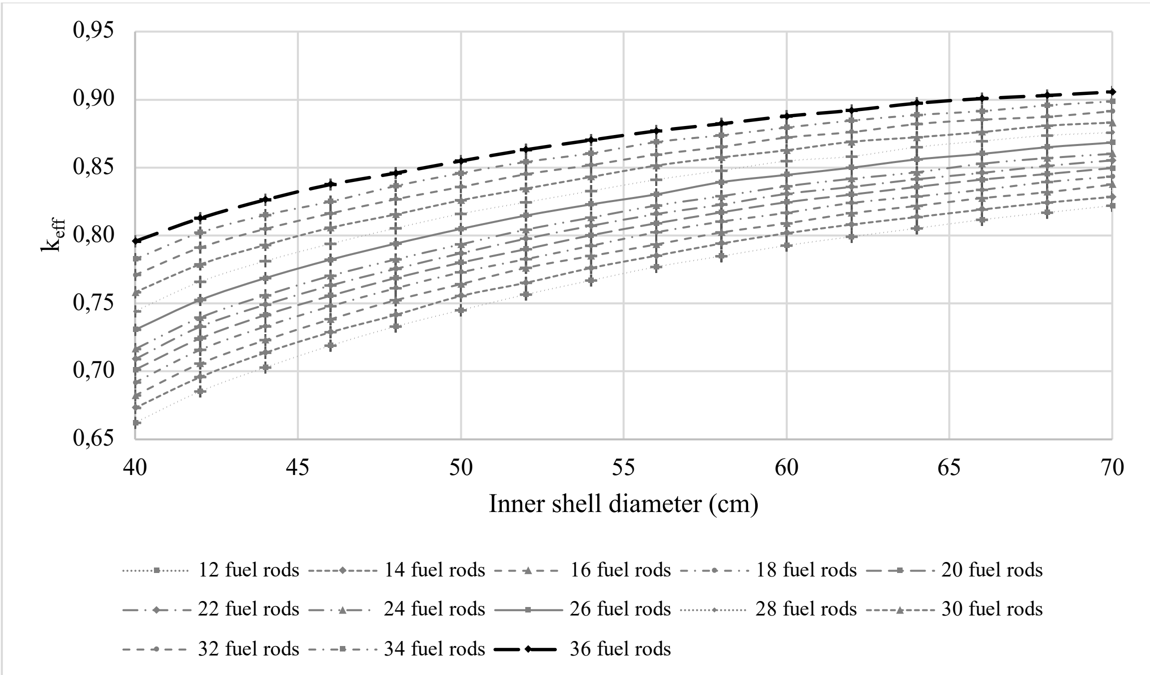


FIGURE 3. Multiplication factor (k_{eff}) as a function of inner shell diameter and number of fuel rods for Design-A.

As seen in **FIGURE 3**, the increasing of inner shell diameter will reduce the neutron leakage so that the greater the reactor dimension, the less neutron to leak. The reduction of neutron leakage will cause an increase in reactor criticality. However, at a certain size, the addition of the reactor dimension no longer has a significant effect in increasing the reactor's criticality. Hence, in this study, the authors also varying the number of fuel rods to increase the ^{235}U mass in the inner shell. More ^{235}U mass means more fissile content thus increasing the thermal utilization factor. But as seen in Fig. 3, the addition of up to 36 fuel rods still does not make the reactor in critical condition.

Then, uranium content is varied in TRIGA fuel to increase the thermal utilization factor. In this study, the inner shell diameter is kept at 70 cm and thorium-uranyl nitrate concentration is kept constant the same as the previous study, the calculation results are shown in **FIGURE 4**.

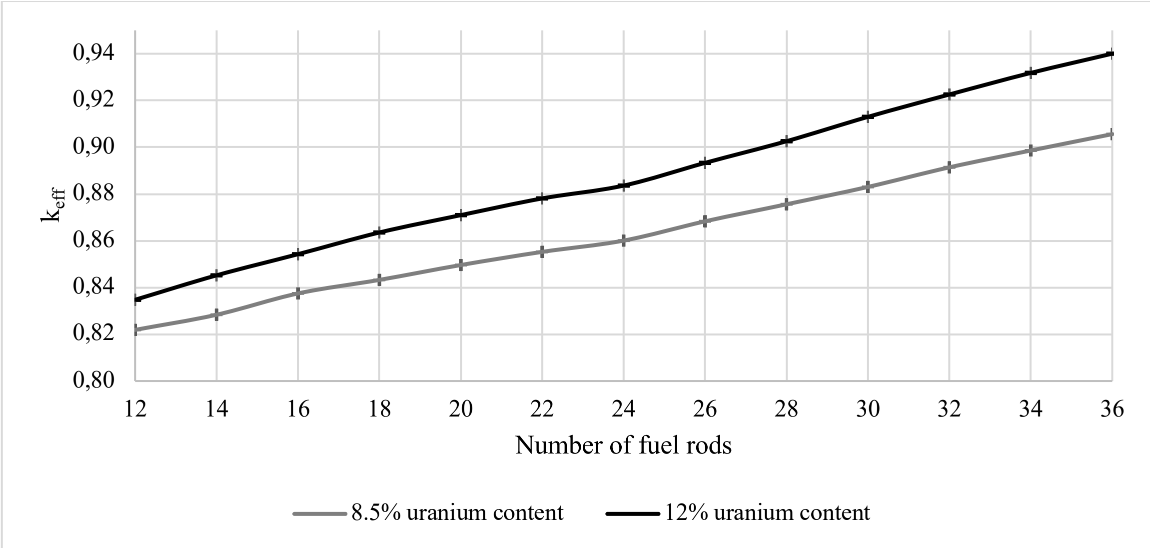


FIGURE 4. Multiplication factor (k_{eff}) as a function of the number of fuel rods and uranium content in TRIGA fuel for Design-A.

As seen from Fig. 4 that the greater on uranium content in TRIGA fuel will increase the k_{eff}. Although using 36 fuel rods it still not enough to achieve a critical state. To achieve the critical mass needed, the study continued by adding fissile content by increasing the concentration of thorium-uranyl nitrate. Increasing concentration of thorium-uranyl nitrate means more fissile content but lower water content in fuel solution. This treatment will increase the thermal utilization factor because there is more fissile content and lower moderator (water) content. But this treatment also reduces resonance escape probability. In the resonance region, the radiative capture cross section of ^{235}U is big enough compared to the fission cross section of ^{235}U so that the neutron energy must be lowered by using a moderator. The calculation results are shown in **FIGURE 5**.

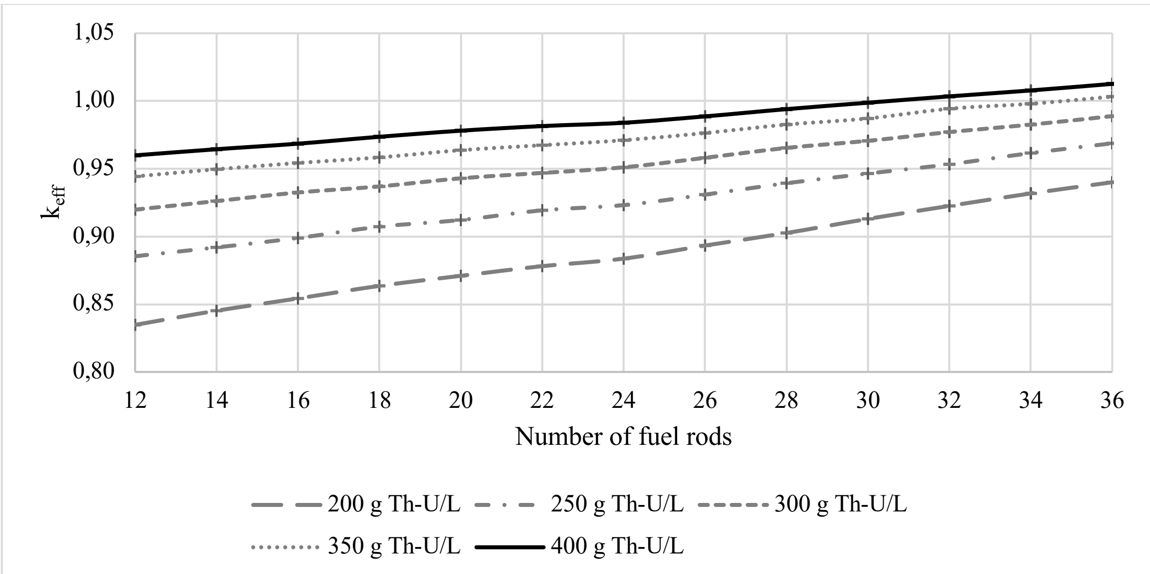


FIGURE 5. Multiplication factor (k_{eff}) as a function of the number of fuel rods and Thorium-Uranyl Nitrate (Th-U) concentration for Design-A.

The addition of thorium-uranyl nitrate concentration will increase the k_{eff}, but at some points it no longer significant in increasing k_{eff}. Moreover, if thorium-uranyl nitrate concentration is very high, it can reduce the k_{eff} because water which acts as a moderator is very small causing the phenomenon of under moderated. From the results above, it can be observed that the reactor can achieve its criticality using thorium-uranyl nitrate with a concentration of 350 g Th-U/L with 36 TRIGA fuel rods uranium content 12% or using thorium-uranyl nitrate with a concentration of 400 g Th-U/L with 32 TRIGA fuel rods uranium content 12%.

Since the critical reactor design has been achieved, then the calculation of CR values was performed using Eq. 4 [19]. The model used for CR calculation is an inner shell diameter of 70 cm with a thorium-uranyl nitrate concentration of 350 g Th-U/L and 36 TRIGA fuel rods uranium content 12%.

$$CR = \frac{3{,}6331\ x\ 10^{12} + 3{,}4793\ x\ 10^{12}}{5{,}8492\ x\ 10^{12} + 3{,}0959\ x\ 10^{13}} \tag{5}$$

$$CR = 0{,}19323 \tag{6}$$

Design-A can achieve its criticality but can't be called a thermal breeder reactor because the CR value is still below 1. In this design, neutron flux in TRIGA fuel is 1,5 times greater than the total flux in thorium-uranyl nitrate fuel. Because the neutron flux in the TRIGA fuel is greater, the collision rate in the TRIGA fuel is also increasing, thus reducing the CR value.

Although the ^{235}U content in thorium-uranyl nitrate is lower (where the weight ratio of Th and U is 60:40) but it has greater fission and radiative capture cross section than the radiative capture cross section of ^{232}Th and ^{238}U (see **FIGURE 6**). CAMOLYP is not able to be a thermal breeder reactor using the current composition of thorium-uranyl nitrate.

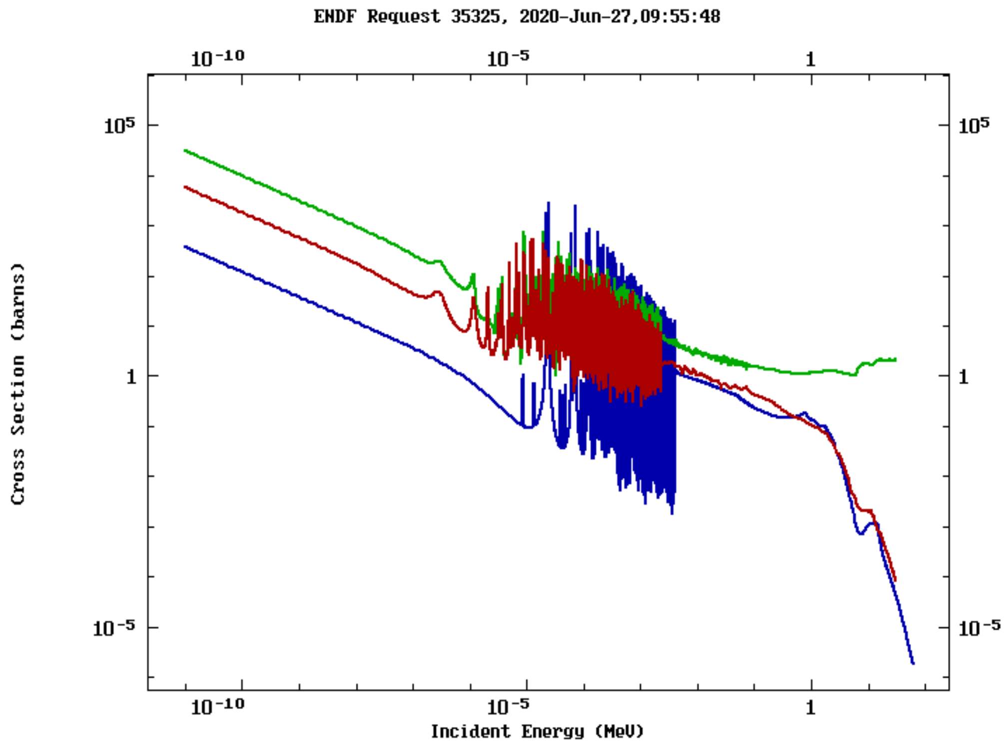


FIGURE 6. Some cross section of material in Thorium-Uranyl Nitrate. The green line is the fission cross section of ^{235}U, the red line is the radiative capture cross section of ^{235}U, and the blue line is the radiative cross section of ^{232}Th [23].

Design-B Analysis

Like in Design-A, the Design-B has also been ensured that the model is in critical condition. In the beginning, the variation of core diameter and uranyl nitrate concentration is carried out in this study. Blanket thickness is kept at 20 cm and thorium nitrate concentration is also kept at 150 g Th/L. The criticality calculation result is shown in **FIGURE 7,** where it can be seen that the critical condition in Design-B can be achieved with a uranyl nitrate concentration of 200 g U/L at a core diameter of 50 cm. Uranyl nitrate with a concentration of 200 g U/L is preferred because less uranium will reduce the uranium collision rate and increase CR values (see Eq. 2).

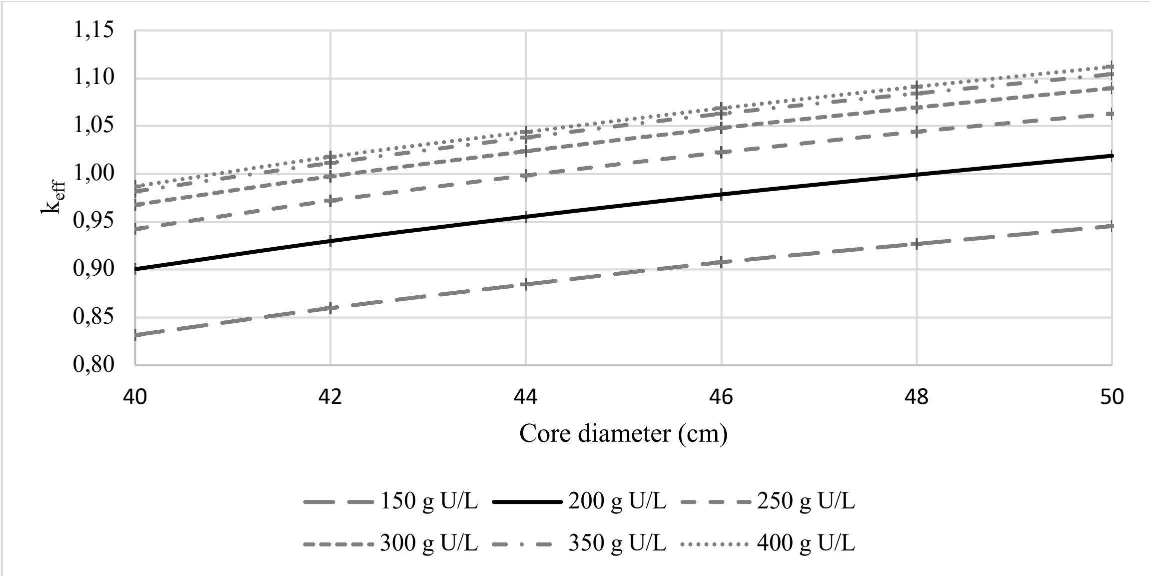


FIGURE 7. Multiplication factor (k_{eff}) as a function of core diameter and Uranyl Nitrate (UN) concentration for Design-B.

Further study is variation in the blanket thickness and thorium nitrate concentration to determine the relationship of the two to the criticality. The study was conducted by varying blanket thickness and thorium nitrate concentration with the reactor core diameter is kept at 50 cm and uranyl nitrate concentration is kept at 200 g U/L. The analysis result is shown in **FIGURE 8**, where it can be concluded that the change in the blanket thickness and thorium nitrate concentration has no significant effect on reactor criticality. Since thorium blanket has a large neutron capture cross section, no matter how big the blanket or thorium nitrate concentration, when neutrons entering the blanket will be absorbed by thorium. If the neutrons pass the blanket region and hit the reflector, it will be reflected to the blanket and has very little chance of the neutron to return to the core. This explains why a change in the blanket thickness and thorium nitrate concentration has no significant effect on reactor criticality.

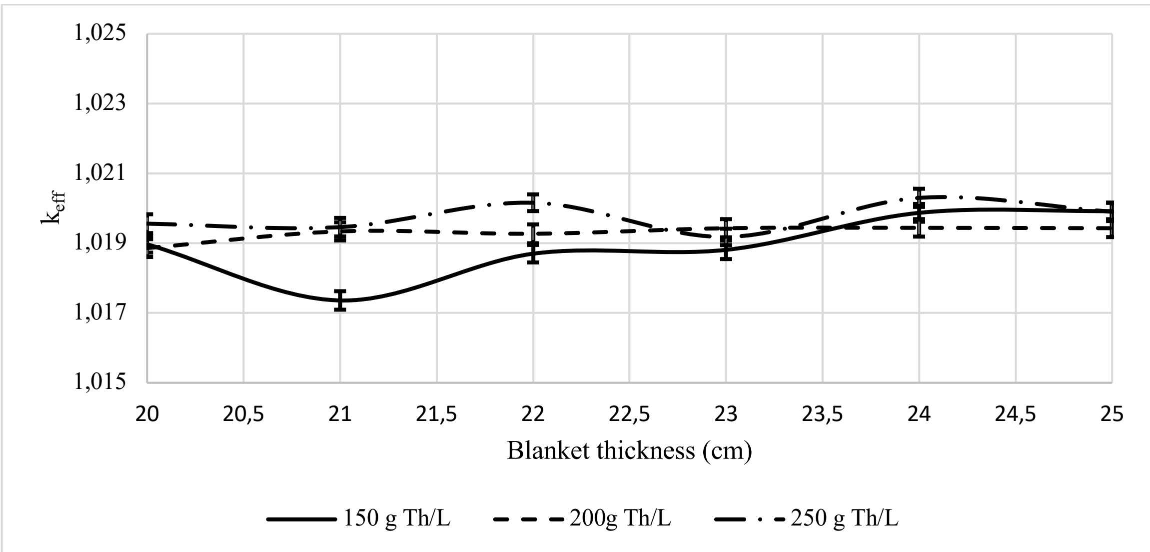


FIGURE 8. Multiplication (k_{eff}) as a function of blanket thickness and thorium nitrate concentration for Design-B.

Furthermore, the blanket thickness was varied with uranyl nitrate concentration kept at 200 g U/L to study its effect on the CR values. The calculation result is shown in **FIGURE 9**, it can be seen that the larger blanket thickness has no significant effect on CR values so the addition of blanket thickness is not a solution to increases the CR value. The number of neutrons entering the blanket remains the same regardless of blanket size. Even the blanket size is enlarged, the number of reactions also remains the same. The number of neutrons entering the blanket can be enlarged by multiplying excess neutrons in the core. Excess neutrons in the core can be increased by reducing the nuclides which have parasitic neutron capture.

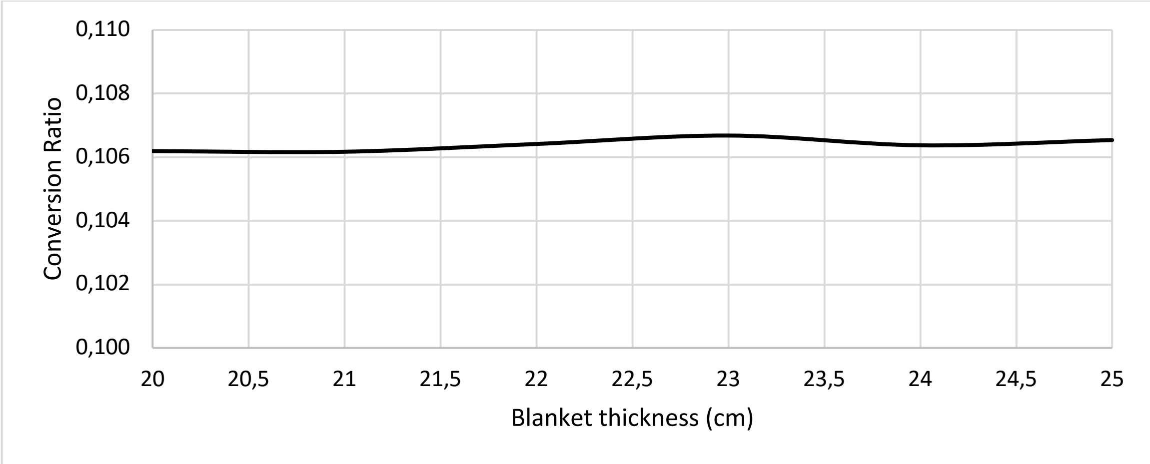


FIGURE 9. Conversion ratio as a function of blanket thickness and core diameter for Design-B.

The concentration of thorium nitrate is also increased to study its effect on CR values. In this study, the core diameter is kept at 50 cm, the blanket thickness is kept at 20 cm, and the uranyl nitrate concentration is kept at 200 g U/L. The analysis result is presented in **TABLE 3** where it can be seen that the increase of thorium nitrate concentration, the more radiative capture (n,gamma) reaction in ^{232}Th. When the concentration of thorium nitrate is increased, there is will be more fertile material (^{232}Th) and less water. Since ^{1}H in water have a large parasitic neutron capture cross section, hence the reduction of water also reduces the nuclides which have parasitic neutron capture. Because nuclides with parasitic neutron capture are reduced, more neutrons will react with ^{232}Th. But the addition of thorium nitrate concentration does not seem to greatly affect the increase in the CR values because when compared to the ^{235}U collision rate on fission and radiative capture, the ^{232}Th collision rate on radiative capture is very small. This is due to the neutron flux in the blanket region is very small that the value only 0,1 time the neutron flux in the core region. Then, it can be concluded that the concept of core and blanket is not effective to increase CR value in the CAMOLYP design because neutron flux on the blanket becomes very small.

TABLE 3. Collision rate and conversion rate for Design-B with core diameter 50 cm, blanket thickness 20 cm, and uranyl nitrate concentration 200 g U/L.

Thorium Nitrate Concentration	Collision Rate (collisions/s)				Conversion Rate
	Uranyl Nitrate			Thorium Nitrate	
	^{235}U		^{238}U	^{232}Th	
	n,gamma	n,fission	n,gamma	n,gamma	
150 g Th/L	5,74E+12	3,10E+13	3,12E+12	7,79E+11	0,106189671
200 g Th/L	5,74E+12	3,10E+13	3,12E+12	9,90E+11	0,112001974
250 g Th/L	5,74E+12	3,10E+13	3,11E+12	1,19E+12	0,117192015
300 g Th/L	5,75E+12	3,10E+13	3,12E+12	1,37E+12	0,122016754
350 g Th/L	5,74E+12	3,10E+13	3,13E+12	1,53E+12	0,126910153
400 g Th/L	5,74E+12	3,10E+13	3,12E+12	1,69E+12	0,13096605

For the reactor to be a breeder, the neutron density must be large. Fast breeder reactor uses ^{239}Pu as fissile material and ^{238}U as fertile material with a fast neutron spectrum. In Fig. 10 we can see that the ^{239}Pu fission cross section is so much higher than the ^{238}U radiative capture cross section. Conversely, in fast spectrum ^{238}U radiative capture cross section is comparable to ^{239}Pu fission cross section. This means the ratio between the collision rate on the fertile material to fissile material can be enlarged which makes the CR value higher. Besides, the value of η (number of neutrons emitted in fission per neutron absorbed in the fissile isotope) is much higher in the fast spectrum [24]. One neutron emitted is used for chain reactions, while the rest is used to bombard fertile material or loss due to leakage or absorbed by other isotopes.

Because CAMOLYP uses a thermal neutron spectrum where the cross section difference between ^{232}Th to ^{235}U is very large and also the neutron emitted by ^{235}U each fission only 2,4 [24], then CAMOLYP need to be designed so that the neutron loss due to leakage or absorbed by other isotopes must be as minimal as possible.

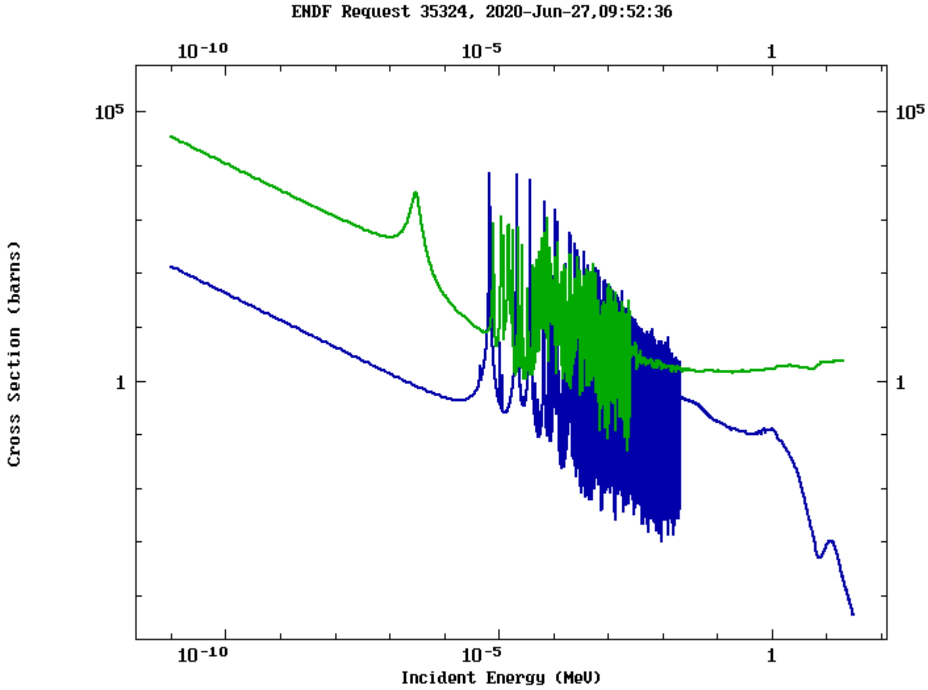


FIGURE 10. Comparison between ^{239}Pu fission cross section (green line) and ^{238}U radiative capture cross section (blue line) [23].

CONCLUSION

The current CAMOLYP design can't be able to become a thermal breeder reactor. The CAMOLYP reactor core with Design-A can achieve its criticality condition with CR of 0,203644. While CAMPOLYP with Design-B also can achieve its criticality with the highest CR of 0,1309661.

Based on the analysis results it can be concluded that to increase k_{eff} value, it can be done by enlarging the inner shell diameter for Design-A and enlarging the core diameter for Design-B. But at some point, enlarging the inner shell or core diameter has no significant effect on increasing k_{eff} value because the neutron leaks from the reactor can't be completely eliminated. Another way to increase k_{eff} value is by increase fissile content by adding ^{235}U content in TRIGA 104 fuel or ^{235}U concentration in thorium-uranyl nitrate solution. Increasing ^{235}U concentration in thorium-uranyl nitrate solution also reduces water content so resonance escape probability will decrease. Therefore, increasing ^{235}U concentration in thorium-uranyl nitrate solution to increase k_{eff} value can only be done up to a certain point.

The value of CR can be increased in a way increasing the concentration of thorium in fuel solution so there will be more reactions between the neutron and thorium. Increasing blanket thickness does not affect CR because the neutrons entering the blanket remains the same.

REFERENCES

[1] M. A. Naafs, "The Global Impact of the Mo-99 Shortage," *Biomed. J. Sci. Tech. Res.*, vol. 4, no. 5, pp. 1–6, 2018, doi: 10.26717/bjstr.2018.04.001114.

[2] A. Youker *et al.*, "A Solution-based approach for Mo-99 production: considerations for nitrate versus sulfate media," *Science and Technology of Nuclear Installations*, 2013, Article ID 402570.

[3] IAEA, Homogeneous Aqueous Solution Nuclear Reactors for the Production of Mo-99 and Other Short-Lived Radioisotopes, IAEA TECDOC Report 1601, Vienna, 2008.

[4] Syarip, T. Sutondo, E. Trijono, and S. Endang, "Design and Development of Subcritical Reactor by Using Aqueous Fuel for Mo-99 Production," *Proceedings of the Pakistan Academy of Sciences: A. Physical and Computational Sciences*, 55 (1): 21–26, 2018.

[5] R. Vega, "Design of a Subcritical Aqueous Target System for Medical Isotope Production," Undergraduate thesis, Texas A&M University, 2014.

[6] B. Delphito and Syarip, "Analysis of Mo-99 production as function of CAMOLYP reactor power," *J. Phys.: Conf. Ser.* 1568 012018, 2020, doi:10.1088/1742-6596/1568/1/012018.

[7] B. Delphito and Syarip, "Neutronic analysis of critical assembly for moly-99 production reactor based on mixed Th-U fuels," *J. Phys.: Conf. Ser*. 1436 012102, 2020.
[8] B. Delphito and Syarip, "Kajian awal analisis neutronik reaktor produksi isotop molly berbasis thorium," (in Indonesian), *Prosiding Seminar Keselamatan Nuklir 2019*, Bandung, 26 August 2019.
[9] G. Zheng *et al.*, "Feasibility study of thorium-fueled molten salt reactor with application in radioisotope production," *Ann. Nucl. Energy*, vol. 135, 2020, doi: 10.1016/j.anucene.2019.106980.
[10] J. R. Maiorino, "An Overview of Thorium Utilization in Nuclear Reactors and Fuel Cycles," *Proceedings of the 12th International Conference of the Croatian Nuclear Society,* Zadar, Croatia, 3-6 June 2018.
[11] A. M. Perry and A. M. Weinberg, "Thermal Breeder Reactors," *Annu. Rev. Nucl. Sci.*, vol. 22, no. 1, pp. 317–354, 1972, doi: 10.1146/annurev.ns.22.120172.001533.
[12] J. C. Clayton, "The Shippingport Presurrized Water Reactor and Light Water Breeder Reactor, WAPD-T-3007," Pennsylvania, 1993.
[13] DOE Research and Development, "Shippingport Operations With the Light Water Breeder Reactor Core, WAPD-TM-1542," Pennsylvania, 1986. doi: DE-AC11-76PN00014.
[14] A. Sagita, "Analisis Neutronik Aqueous Homogeneous Reactor (AHR) Reaktor Subkritis Breeder untuk Memproduksi Mo-99 dengan Bahan Bakar Thorinil Sulfate (ThO2(SO4))," (in Indonesian), Bachelor thesis, Universitas Gadjah Mada, 2016.
[15] Syarip, E. Togatorop, "Molybdenum-99 production calculation analysis of SAMOP reactor based on thorium nitrate fuel," *Journal of Physics: Conference Series,* 978 (1), 012072, 2018.
[16] F. Yassar, A. W. Harto, and Syarip, "Neutronic analysis of subcritical assembly for 99Mo isotope production based on thorium nitrate fuel," *J. Phys.: Conf. Ser.*, 1080 012021, 2018.
[17] D. B. Pelowitz, *MCNPX User's Manual* (Los Alamos National Laboratory, Oak Ridge, 2011).
[18] X-5 Monte Carlo Team, *MCNP – A General Monte Carlo N-Particle Transport Code, Version 5 Volume I: Overview and Theory* (Los Alamos National Laboratory, Oak Ridge, 2008).
[19] S. Permana, N. Takaki, and H. Sekimoto, "Breeding and void reactivity analysis on heavy metal closed-cycle water cooled thorium reactor," *Ann. Nucl. Energy*, vol. 38, no. 2–3, pp. 337–347, 2011, doi: 10.1016/j.anucene.2010.10.009.
[20] M. B. Chadwick *et al.*, "ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data," *Nucl. Data Sheets*, vol. 112, no. 12, pp. 2887–2996, 2011, doi: 10.1016/j.nds.2011.11.002.
[21] H. Suwarno, "Development of TRIGA fuel fabrication by powder technique," *Atom Indonesia*, vol. 40, no. 3, pp. 113–119, 2014, doi: 10.17146/aij.2014.329.
[22] T. Zagar and M. Ravnik, "Fuel Element Burnup Determination in HEU - LEU Mixed TRIGA Research Reactor Core," *International Meeting on Reduced Enrichment for Research*
[23] IAEA, "Evaluated Nuclear Data File (ENDF)," 2020. https://www-nds.iaea.org/exfor/endf.htm (accessed 1 September 2020).
[24] J. R. Lamarsh, Introduction to Nuclear Reactor Theory (John Wiley & Sons, New York, 1978).

Parameter Analysis of Triga Research Reactor with PCTRAN Pool Reactor Simulator

Rima Walhikmah[1, a)] and Rida SN Mahmudah[2, b)]

[1]Universitas Negeri Yogyakarta. Jalan Colombo No.1 Daerah Istimewa Yogyakarta 55281, Indonesia
[2]Universitas Negeri Yogyakarta. Jalan Colombo No.1 Daerah Istimewa Yogyakarta 55281, Indonesia

[a)] rima2255fmipa@student.uny.ac.id
[b)] rida@uny.ac.id

Abstract. TRIGA is one of research reactor owned by Indonesian government and managed by National Nuclear Energy Agency (BATAN). This study aimed to simulate and analyze several reactor's condition in its normal and malfunction operation. The simulation was carried out by using PCTRAN Pool Reactor, a personal computer simulator software distributed by IAEA for training and education purposes. This simulator has simplified neutronic and thermal hydraulic model of TRIGA reactor, as well as its safety system. The simulation results show that at startup, the fuel temperature was approximately 300ºC and decreasing over time to room temperature accompanied by big changes in resulted neutron flux. At normal operation, the simulator can produce a steady critical condition, in correspond with its reactivity. Meanwhile, at transient condition, the center fuel temperature increased rapidly to 350ºC due to the loss of coolant in the pool, and an inherent safety system was visible during simulation. Despite of its simplicity, this simulator showed an important aspect of reactor's neutronic parameters.

Keyword: TRIGA Reactor, PCTRAN, Simulation reactor, inherent safety

INTRODUCTION

Nuclear reaction, especially nuclear chain reaction can be greatly of advantage in human lives. However, our society still see nuclear in negative way. Nagasaki and Hiroshima tragedy in 1945 and Chernobyl accident in 1986 are two of several reasons of this. Whereas, with proper technology, nuclear can contribute to advancing science and technology, as well as advancing human prosperity.

One of technologies utilizing nuclear reaction is nuclear reactor. Nuclear reactor is a place where nuclear chain reaction occurs. To control and stop nuclear operation, it has a neutron absorber material called control rod.

Nuclear reactors are classified based on their purposes and utilities, neutron energy and their components as well as their operational parameters. Based on neutron energy undergo fission, reactors are classified into fast and thermal reactors. Based on their parameters, reactors are classified to graphite reflected reactors, light water cooled reactors, and high temperature reactors. Meanwhile, based on their purposes and utilities, reactors are classified into power reactors, research reactors for material testing and training, and isotope production reactors that are usually categorized as research reactors [1].

Research reactors are nuclear reactors for research, material testing, education/training as well as radioisotope production. Power reactors are nuclear reactors FIGUREd to produce electricity or power plant. The difference between these two is, on research reactors, their main utilization is neutrons from fission reactions which are used to do research and isotope production. Resulted heat is FIGURE to be as low as possible, so it can be released to the environment. In power reactors, their main utilization is high temperature and pressure vapor, heated by energy released from fission reaction to power turbine and generate electricity. Neutrons resulted from fission are either absorbed or ignored to maintain chain reactions [2].

One of established research reactors is TRIGA (Training, Research, Isotopes, General Atomics) which was figured and built in small call by General Atomics, as the name implies. TRIGA is a pool type reactor that can be built without containment and can be used by scientific institutions and universities for research, undergraduate and graduate studies, companies, non-destructive test and isotope production [3].

According to IAEA (International Atomic Energy Agency), there are 33 TRIGA reactors in the world. Those reactors are spread in several countries such as Austria, Bangladesh, Indonesia, Italy, Malaysia, Mexico, Romania, Slovenia, Japan, United State of America, Germany and England. From those 33, 17 is on operation, 3 is under decommissioned and other 13 are in nonactive status [4].

Indonesia has three research reactors operated and maintained by BATAN. Those reactors are Multipurpose reactor GA Siwabessy with 30 MW power located in Puspiptek Serpong Tangerang, TRIGA Mark II with 2 MW power located in Bandung, and TRIGA Mark II Kartini with 100 kW power located in Yogyakarta.

As the name implies, TRIGA reactors can be used not only for isotope production and irradiation, but also for training and education. These include training for officer and supervisor of reactor operator, reactor maintenance, radiation protection, radioisotope processing, neutron activation analysis, as well as neutronic and thermal hydraulic experts, radiochemistry and radiopharmacy [5].

Safety and security aspects are important things to consider in operating TRIGA and other reactors. One of parameters need to be observed is reactivity changes due to temperature. Fuel temperature changes will lead to core temperature changes. This parameter called reactivity coefficient, consists of fuel reactivity coefficient (Doppler effect) and moderator reactivity coefficient [6].

One of requirement needed in a reactor according to IAEA is the inherent safety feature, ISF. This feature will differ in one reactor and another and can avoid power deviation that can lead to a shutdown—such as in Chernobyl accident [7].

In addition, it is also necessary to understand the default condition such as startup and shutdown system, emergency system and scram condition of the reactor to operate it safely. To help understand those conditions, a reactor simulator such as PCTRAN can be used.

PCTRAN is one of computer software used to simulate nuclear reactor developed by MST (Micro-Simulation Technology) and used by IAEA. It can simulate various normal operation as well as emergency situation. The advantages of PCTRAN are: it can be installed in a personal computer, and can operate faster than the real time [8]. There are at least four reactor simulator used by IAEA for education and research, i.e. Pressurized Water Reactor (PWR) Simulator, Boiling Water Reactor (BWR) Simulator, Pressurized Heavy Water Reactor (PHWR) and TRIGA Pool Experimental Reactor Simulator.

This study used PCTRAN Pool Reactor or TRIGA Pool Experimental Reactor Simulator. In this pool-type TRIGA, there are two power capacity options to be chose before running the simulator, i.e. 250 KW and 1 MW.

A. S. Mollah, et. al, reviewed this software in their paper "PCTRAN: Education Tool for Simulation of Safety and Transient Analysis of a Pressurized Water Reactor" [8]. It discussed several simulation results of various operating and transient condition as described in PCTRAN's module provided by IAEA. This paper showed that at the first 200 second, decay heat is disappearing and only discussed initial condition after reactor shutdown.

This study aimed to analyze other operating conditions, such as at normal operation and when malfunction occurs. At normal condition, we observed and analyzed the reactor's start up and scram, when all of the control rods are inserted by force. While malfunction condition was simulated when water level in the core is loss by 50%.

Reactor start up can run smoothly if all parameters are in normal level and all procedures done by following IAEA guidance. If one of parameters is fail, then reactor will be on malfunction condition. Some of those parameters are fuel temperature, control rod position, reactivity, neutron flux and reactor power.

LITERATURE REVIEW

TRIGA Pool-type Experimental Reactor

TRIGA reactor has 250 kW power with neutron flux of $10^{13} n/cm^2/s$. The reactor's specifications used in this simulation are:

- Reactor's fuel is cylinder type contains uranium-zirconium hydride (UZRH). UZRH is fissile material with 19% U-235 enriched in fuel assembly.
- Has 4 control rods made of boron, utilized as neutron absorber. These control rods are functioned as shim rod, safety rod and regulating rod.
- Graphite as moderator [5]
- Reactor pool containing coolant water. There are two layers of water in the pool, i.e. warm layer and deep layer. The water in warm layer is used to detain active impurity flow from reactor core to the pool and to maintain radiation level in the pool surface in operation range. In addition, warm water system also controls temperature to make it higher than the water below (in the deep layer) [9].

PCTRan Pool Reactor

PCTRan Pool Reactor is one of the reactor simulators provided by MST (Micro-Simulation Technology) and has been used by IAEA for education purposes. This software used point kinetic model with 6 group of delayed neutron. This model includes simulation of Iodine-135 – Xenon-135 poison decay chain produced from fission, with “fast time” ability [5].

There are several easy-to-use features in this Pool Reactor software, such as:

1. In the right bottom corner, there is time display to control running process, such as freeze, resume (any real time), snap (simulator current condition is saved in a hard disk file and can be used later as initial condition), reset (change the current condition back to initial-saved condition), slow time (to slow down simulation), and fast time (to fasten simulation).
2. User can easily know the neutron flux of various condition, by using the following features: source range (neutron flux lower than $10^4\ n$ equal to $10^{-3}\%$), intermediate range (neutron flux between $10^3 - 10^9\ n$), and power range (neutron flux between $10^7 - 2\ x\ 10^9\ n$).
3. Manual “scram” button.
4. Feature to lower down and raise up the control rods with constant rate.

FIGURE 1 shows the front display of PCTRAN Pool Reactor.

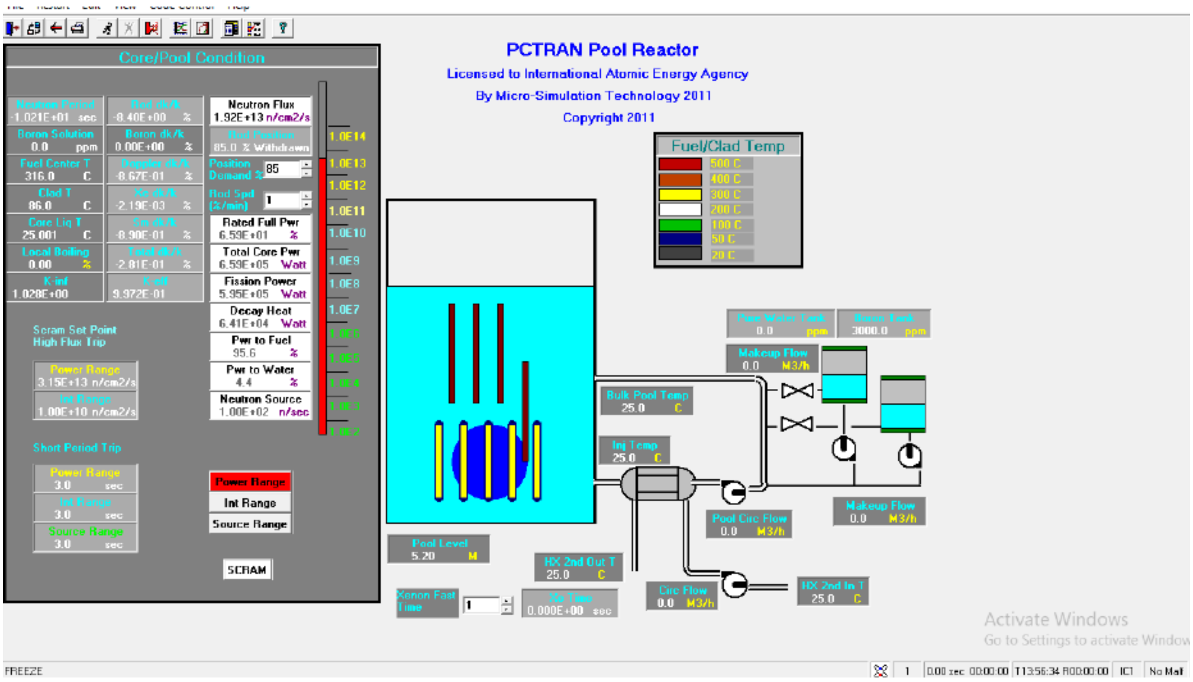


FIGURE 1. PCTRAN Pool Reactor Display

6 Groups Point Kinetic Model

Kinetic model with 6 groups of neutron delayed energy and the reactivity control are defined as:

$$\frac{dn}{dt} = \frac{\rho - \beta}{t} n \sum_{i=1}^{6} \lambda_i C_i + S \qquad (1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{t} n - \lambda_i C_i \qquad (2)$$

where n is neutron density, ρ is reactivity, defined as $(k-1)/k$, with k is effective multiplication factor, β_i is delayed neutron fraction of i-th group, t is neutron change time, λ_i is decay constant of i-th group, C_i is precursor nuclide concentration of i-th group and S is the neutron source.

In this simulator, kinetic model in Eq. (1) is solved numerically by using finite difference method. Meanwhile, the reactivity is governed by the control rod motion and the adjustment of boron concentration.

The simulator uses reactivity correction due to Xe-135 poisoning and other influencing nuclides as described in [8].

Pool Coolant System

PCTRAN/Pool Rx uses semi empirical methods to model pool circulation as its pool coolant system. Temperature circulation for desludging and disposal is formulated as:

$$T_H - T_C = \frac{Q_{HX}}{W_{HX} Cp} \quad (3)$$

where Q_{HX} is the heat removal rate from the heat exchanger. Its value is equal to the core power Q_{core} in the steady state condition. $T_H\ dan\ T_C$ is the hot and cold temperature, or the output and input temperature of the pool.

In the transient condition, bulk temperature of the water pool is defined as:

$$\left(MC_{p,water}\right) \frac{dT_{pool}}{dt} = Q_{core} - Q_{HX} \quad (4)$$

where M is the water's mass.

METHODOLOGY

As mentioned above, this study simulated TRIGA reactor in various conditions using PCTRAN Pool Reactor. We chose TRIGA power of 250 kW in all conditions. For the initial condition (IC), there are 9 options available in the simulator, i.e.:

- IC 1: steady state with 74% power
- IC 2: Shutdown all rods
- IC 3: Max Doppler
- IC 4: 3% Power Max. Mod. Temperature
- IC 5: 1E6 neutron source 8% rod
- IC 6: 250 KW 100% Power
- IC 7: Shutdown ININ
- IC 8: Shutdown Time=0
- IC 9: ININ TRIGA 100% Power

Several properties of TRIGA used in this simulator are shown in **TABLE 1**.

TABLE 1. Reactor Properties

Parameter	Value
Neutron lifetime (sec)	3.8461×10^{-5}
Pool cross section area (m²)	4.00
Effective delay neutron fraction	6.5×10^{-3}
Borid Acid tank boron concentration (ppm)	3×10^{3}
Rated neutron flux at full power (n/cm²/s)	1.013×10^{13}
Fuel length (m)	7×10^{-1}
Initial pool water level (m)	5.2
Rated thermal power per core (MW)	2.5×10^{-1}
Initial building pressure (bar)	1.03
Initial average fuel temperature (°C)	2.0×10^{2}
Initial building temperature (°C)	2.5×10^{1}
Building total volume (m³)	1.0×10^{4}
Pool pump flow rate (m³/hr)	7.7×10^{1}

In this study, we simulated the following conditions:

Start up

To begin this condition, we can choose inherent initial condition IC1 from the menu or we can do it manually. To do the second option, we set the IC available at "restart" menu. Fill IC's number with unused number (in this study, we used number 10) and give an appropriate description (example: start up). In this condition, the rod position = 0, meaning that all control rods are in the reactor core.

To reach normal condition, change control rod position to 85% and set its rising rate to 20%/minute. This means, that the rod will be withdrawn from the core until only 15% of it remaining in the core with 20% of raise per minute.

Fuel temperature profile in the start up condition can be seen in Fig. 2. At first, fuel temperature was 300˚C but due to the 0% of the control rod position, there was no fission reaction in the core. Thus, the fuel temperature decreased rapidly due to the cooling pool. Slowly, as the control rod being raised, the fuel temperature is also increasing. It takes approximately 300 second to reach 85% rod position, and the fuel temperature begin to rise at approximately 450 second. We can see that at steady state, fuel temperature didn't reach its initial level because the control rod was not withdrawn completely from the core.

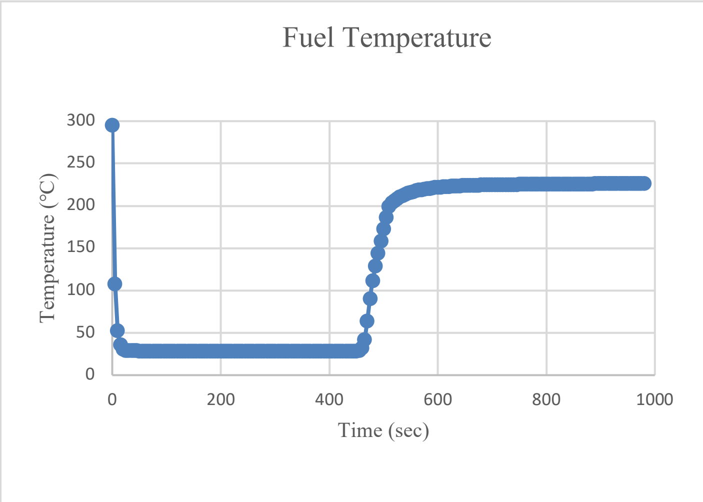


FIGURE 2. Fuel temperature profile at start up

Normal Condition

Normal condition is defined as reactor operating normally without any malfunction. To simulate this, we can choose IC 1 available in the menu. In this condition, we analyze the reactor's criticality.

Effective multiplication factor, k_{eff}, or simply k, is an important parameter in analyzing reactor's criticality. It is defined by the following equation [8]:

$$k = \frac{N_{(t=t_i)}}{N_{(t=t_{i-1})}} \tag{5}$$

where:

$N_{(t=t_i)}$= neutron population at i-th generation

$N_{(t=t_{i-1})}$= neutron population at previous generation

Based on its multiplication factor, there are 3 types of reactor condition, i.e.: a) $k > 1$, reactor is in supercritical condition, meaning that neutron population is increasing; b) $k = 1$, reactor is in critical condition, meaning that neutron population is not changing; c) $k < 1$, reactor is in subcritical condition, meaning that neutron population is decreasing.

Loss of Pool Water Reactor (LOPWR)

LOPWR is a condition when the water level of the pool decrease or even loss. In the normal condition, the water height in the pool is 5.20 m (100%). In this study, the level was lowered to 2.60 m (50%) and 60 cm. We chose 60 cm level water, lower than the height of fuel length (see **TABLE 1**) to investigate the simulator's capability in imitating TRIGA's inherent safety.

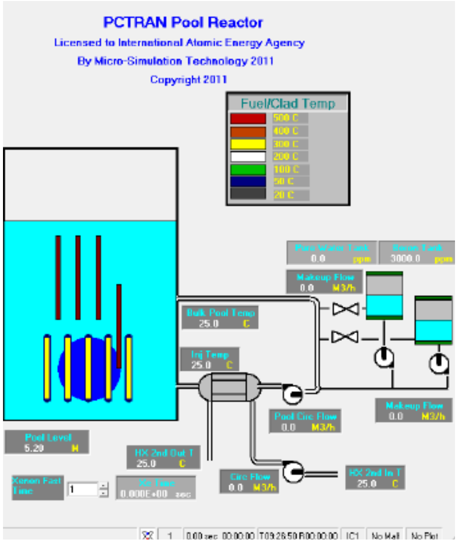


FIGURE 3. 100% water level in the core

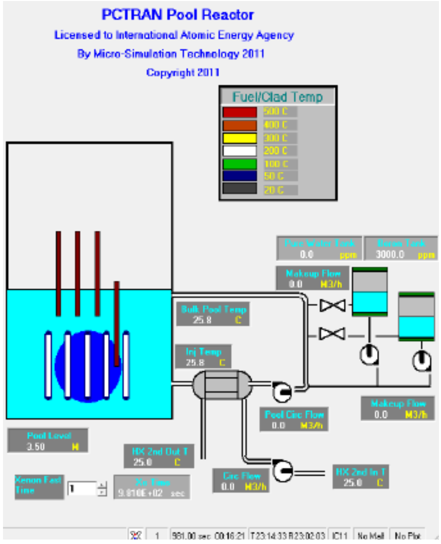


FIGURE 4. Lowered water level

To simulate LOPWR, we chose a normal initial condition. Then, the water level is lowered by changing "pool water level" in the basic data in the edit menu. The display of full and lowered water level can be seen in **FIGURE 3** and **4**.

We simulated above 3 conditions and analyzed various parameters change during the simulation. In the next part of this paper, we show graphs of the real time simulation results, even though the simulation was carried out with fast-time modes.

RESULTS AND DISCUSSION

Start up

Start up is defined by starting moment of the reactor, since 0% power until it reaches the steady state of 100% power. It is important to make sure that several parameters are in the safe level according to BAPETEN, during this phase.

As mentioned above, at start up we changed the control rod position from 0% to 85% with the rise rate of 20% per minute. During this process, heat exchanger is not operating or being turned off.

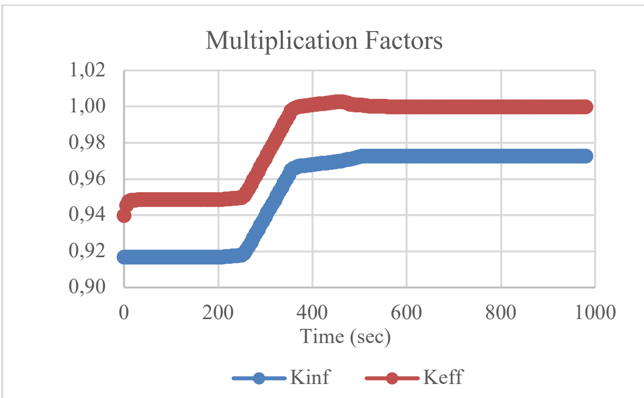


FIGURE 5. Multiplication factors at start up

From the simulation results as shown **FIGURE 5,** it can be seen that the neutron flux can reach power range in less than 4 minutes. When k_{eff} approaching 1.0 (critical), the neutron flux switched to intermediate range.

When the control rods are rising, effective multiplication factor k_{eff} is also increasing. This is because, the rising of control rods means less boron in the pool, thus less neutrons are being absorbed. This leads to more neutron in the next generation, hence the increasing k_{eff}. After 300 second, k_{eff} began to stabilize and being critical around 500 second.

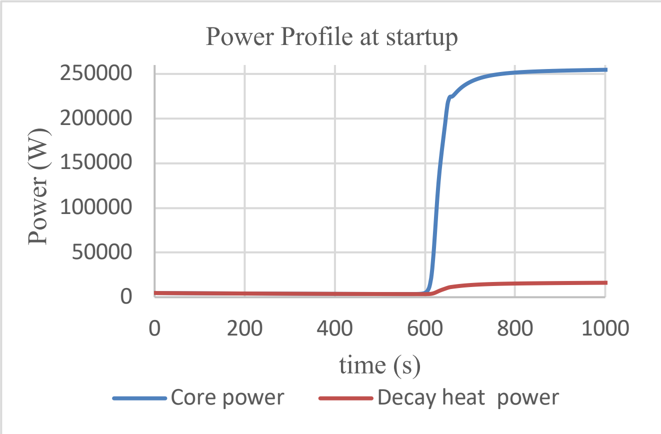


FIGURE 6. Power profile at start up

As can be seen from **FIGURE 6**, the reactor core power remains 0 and start increasing at 600 second, i.e. when the k_{eff} becomes critical. It finally reaches 100% power, 250 kW, at approximately 750 second after start up. As the core power increasing, the decay heat power is also increasing. This is due to the decay of fission products, occurring after the core capable to maintain fission chain reaction.

Normal Condition

As mentioned above, without any malfunction, the reactor can operate normally, shown by a smooth running in the simulator. **FIGURE 7** and **8** show k_{eff} and reactivity profiles at normal condition, respectively.

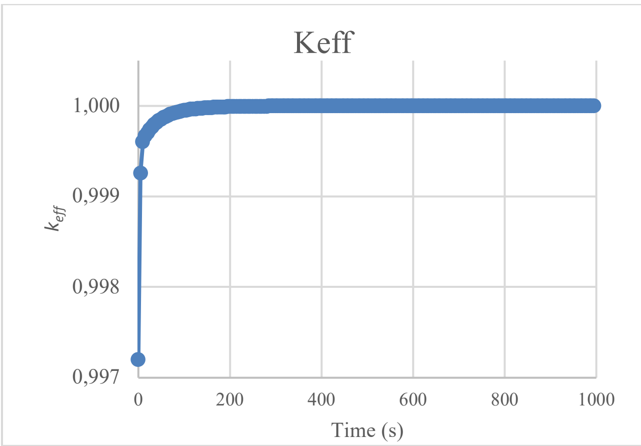


FIGURE 7. k_{eff} profile at normal operation

From **FIGURE 7** above, we can see that reactor reach critical at 360 second. Meanwhile, at the same time, reactivity approaches 0. We also can see that after k_{eff} steady and being critical, reactivity is also steady being 0.

Nuclear reactors have inherent factors that can change reactivity even those designed to operate in constant power. There are several factors contributing to reactivity change, such as xenon concentration, fuel amount in the core, or void in the moderator. Those factors are stated in reactivity coefficient unit.

Reactivity coefficient of temperature is defined as partial differentiation of reactivity against temperature. Reactors that have a positive reactivity coefficient of temperature, reactivity will increase if temperature increase. On the other hand, those have a negative reactivity coefficient of temperature will experience decrease reactivity when the temperature increase. Most of reactors have a negative reactivity, due to the Doppler effect. This effect strongly affects the reactor's safety during its operation, but beyond the scope of this study.

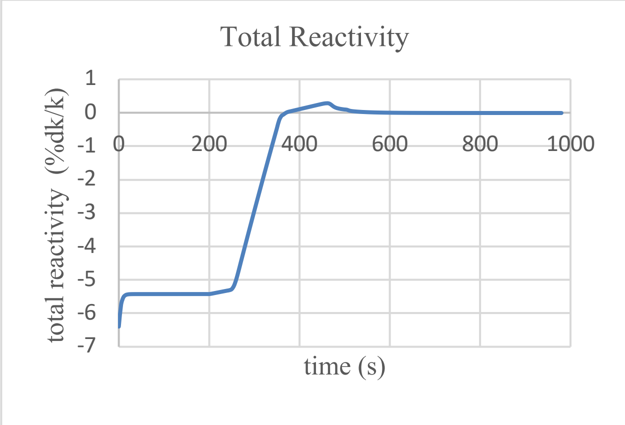


FIGURE 8. Total reactivity profile

Loss of Pool Water Reactor

In this part, we reduce the reactor's pool water level to see the effect of coolant loss to the reactor system as well as the simulator's inherent safety feature. **FIGURE 9** shows the center line fuel temperature at three different water levels.

At start up, the average fuel temperature was 200˚C (see **TABLE 1**), with the hottest part in the center (approximately 315˚C). The first 100 second after operation, the center fuel temperature was decreasing rapidly because the reactor still in a subcritical condition. After it pass the critical condition, i.e. it can maintain fission chain reaction, the center fuel temperature began to increase.

As we can see in **FIGURE 9**, the center fuel temperature with 2.6 m of water level is higher than that of 5.2 m. This is due to the difference of coolant volume in the pool. The lower the water level, then the coolant volume is also smaller, thus reduce the core's ability to cool the fuel down.

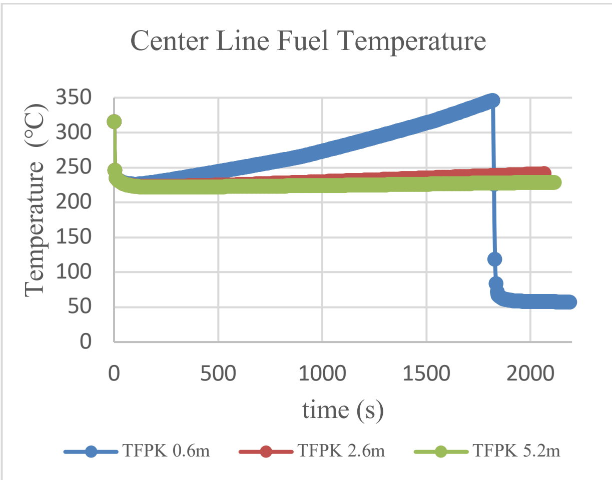


FIGURE 9. Fuel temperature profile of different pool water levels

We also can see that a significant center fuel temperature increase occurs when the water level was reduced to 0.6 m. As mentioned above, this is lower than the height of fuel length by 10 cm. Meaning that 10 cm of fuel rod does not covered by cooling water. Apart from a big reduce of coolant volume, this also leads to absolutely no cooling process experienced by 1/7 of the fuel. Thus, its temperature was increasing rapidly, followed by an automatically scram at $t = 1820$ second. The control rod position directly changed to 0%, meaning all of them are inserted in the core and stopped the fission reaction. This leads to a rapid decrease of center fuel temperature, from approximately 350˚C to a steady state of 50˚C, as we can see in **FIGURE 9**. This confirms that the simulator can imitate a simple inherent safety of TRIGA reactor.

CONCLUSION

In this study, we simulated several conditions of TRIGA operation with PCTRAN simulator. At start up, we observed and analyzed several parameters change during the increasing of power from 0% to 100% (250 kW). The simulation results show that at startup, the temperature was approximately 300ºC and decreasing over time to room temperature accompanied by big changes in resulted neutron flux. At normal operation, the simulator can produce a steady critical condition, in correspond with its reactivity. Meanwhile, at transient condition, the center fuel temperature increased rapidly to 350ºC due to the loss of coolant in the pool, and an inherent safety system was visible during simulation. Despite of its simplicity, this simulator showed an important aspect of reactor's neutronic parameters.

ACKNOWLEDGMENT

This study can be conducted owing to International Nuclear Energy Agency who has provided the nuclear reactor simulators for education and training used in this study

REFERENCES

[1] BATAN. Reaktor dan Klasifikasinya (2020) pada laman http://www.batan.go.id/index.php/id/infonuklir/reaktor-infonuklir/814-
[2] Adiwardojo, dkk. Mengenal Reaktor Nuklir dan Manfaatnya (2015), diakses pada laman http://www.batan.go.id/index.php/id/publikasi-2/bukunuklir?download=7:mengenal-reaktor-nuklir-dan-manfaatnya
[3] L.C. Po. PCTRAN – *Personal Computer Transient Analyzer for a Two-loop PWR And TRIGA Reactor. Workshop on Nuclear Power Simulator for Education Politecnico* di Milano, Itali. (2011)
[4] IAEA. Reactor List (2009), diakses pada laman https://nucleus.iaea.org/RRDB/RR/ReactorSearch.aspx
[5] BATAN. Diusia ke-54 Tahun, Reaktor TRIGA 2000 Tetap Beroperasi dengan Baik. (2019), diakses pada laman http://www.batan.go.id/index.php/id/pstnt-id/beriat-pstnt/5259-di-usia-ke-54-tahun-reaktor-triga-2000-tetap-beroperasi-dengan-baik.
[6] Safarzadeh, O., Derakhshandeh, F. S., & Shirani, A. S. Calculation of reactivity coefficients with burn-up changes for VVER1000 reactor. Progress in Nuclear Energy, (2015), 81, 217-22.
[7] IAEA. "FIGURE Measures to Facilitate Implementation of Safeguards at Future Water Cooled Nuclear Power Plants", Technical Report Series No.392, International Atomic Energy Agency, Vienna. (1998)
[8] A.S. Mollah. *PCTRAN: Education Tool for Simulation of Safety and Transient Analysis of a Pressurized Water Reactor.* Dalam International Journal of Integrated Science & Techonolgy (2018), 3 1-10 ISSN 2411997.
[9] ORNL. Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and FIGURE; Oak Ridge: Oak Ridge National Laboratory. (2011)
[10] BAPETEN. Dokumen Peraturan Kepala Badan Pengawas Tenaga Nuklir tentang Batasan dan Kondisi Operator Reactor Nondaya, diakses pada laman https://jdih.bapeten.go.id/unggah/dokumen/peraturan/218-3_(PERATURAN)-1557814064.pdf

Void Reactivity Coefficient Analysis for Safety of TMSR-500 Using MCNP6

Atika Nur Mufidah[1], Suharyana[1], Riyatun[1], and Azizul Khakim[2]

[1]*Department of Physics, Faculty of Mathematics and Natural Science, Sebelas Maret University, Surakarta, Indonesia*
[2]*Nuclear Energy Regulatory Agency of Indonesia (BAPETEN)*
Gajah Mada Street No. 8 Central Jakarta

Corresponding author: suharyana61@staff.uns.ac.id

Abstract. *Thorium Molten Salt Reactor* (TMSR-500) is one of the developments of the generation IV reactor which has various advantages over the type of reactor in the previous generation because it can produce high resources and long lasting. This reactor uses liquid salt as fuel as well as coolers and moderators made from graphite. The main fuel used is $NaF\text{-}BeF_2\text{-}ThF_4\text{-}UF_4$. Void reactivity is a very important parameter to calculate the safety level of the reactor. Voids can be formed due to heating the fuel to the boiling point of the fuel which causes bubbles to occur in the fuel and decreases the fuel density. This reported study simulates and calculates the void reactivity coefficient due to a decrease in fuel density expressed by the void fraction. Simulations are carried out using the MCNP6 when voids have not formed (0%), formed 10%, 20%, 30%, and 40%. The results of this study indicate that the TMSR-500 is a reactor that operates in the thermal energy spectrum with a temperature of 977K. The criticality of TMSR-500 was obtained with the value of k_{eff} = (1.01804±0.00008), and the void reactivity coefficient was positive at (0.068±0.003) %Δk/k/%void due to the undermoderated condition of the reactor so it was safe.

Keyword: TMSR-500, MCNP6, thermal spectrum, void reactivity coefficient, safety

INTRODUCTION

Molten Salt Reactor (MSR) or liquid salt-fueled reactors, is one of the resource technologies that are suiTABLE to be a renewable energy solution. *Thorium Molten Salt Reactor* (TMSR-500) is one of the generation IV reactors which has various advantages over the type of reactor in the previous generation because it can produce high resources, long-lasting, but lower costs. This reactor operates at high temperatures, but low pressure approaches atmospheric pressure [1-2].

The power produced by the TMSR-500 is 1000 MWe which is divided into 4 modules (in the form of pots) with 250Mwe each. The pot module is one of the 3 core components of the TMSR-500, the other two are the primary heat exchanger and the primary strand pump. Each module in it has a reactor that can be replaced in a closed silo can. Can contain a moderator made from graphite and salt fuel with a mixture of NaF, BeF2, ThF4, UF4 [3]. In this study the percentage of moles used for each fuel component was 76% / 12% / 9.8% / 2.2% with 19.75% enrichment of U^{235} [4]. In the liquid salt fuel, there is graphite that plays a very important role as moderators for thorium-based nuclear reactors.

One aspect of safety for nuclear reactors is void reactivity. Voids in liquid salt reactors occur due to reactor operations whose temperature is close to saturation temperature, causing bubbles to form on the reactor core. The saturation temperature is the temperature at which the vapor pressure of a liquid is equal to the pressure of the environment surrounding the liquid [5]. Voids in MSR can also occur due to the formation of fission products in the gas phase. The formation of voids also causes a reduction in fuel density. The higher the fuel temperature, the higher the voids in the core. The presence of voids will affect the performance of the reactor, this condition can be seen from the estimated value of the criticality. The reactor's criticality is expressed in terms of the magnitude of $k_{eff.}$

This study aims to obtain the void reactivity coefficient value of the TMSR-500 as one of the aspects of reactor safety. The method used is a variation of the void fraction assumed with a variation of the reduction in fuel density by 0%, 10%, 20%, 30%, and 40%. The calculations are carried out using the MCNP6 program which works with the Monte Carlo method.

The value of the void reactivity coefficient can be positive or negative depending on the nuclear system overmoderated or undermoderated. If undermoderated, the results of the void reactivity coefficient are positive, and vice versa [6].

METHODOLOGY

In this study, the core design of the TMSR-500 is based on a model from Devanney [4] using Vised software and version 6 of the MCNP (Monte Carlo N-Particle) program for the running process of k_{eff} calculation. The TMSR-500 core consists of pots, shields, reflectors, moderators, fuels, and control rods which can be seen in **FIGURE 1**.

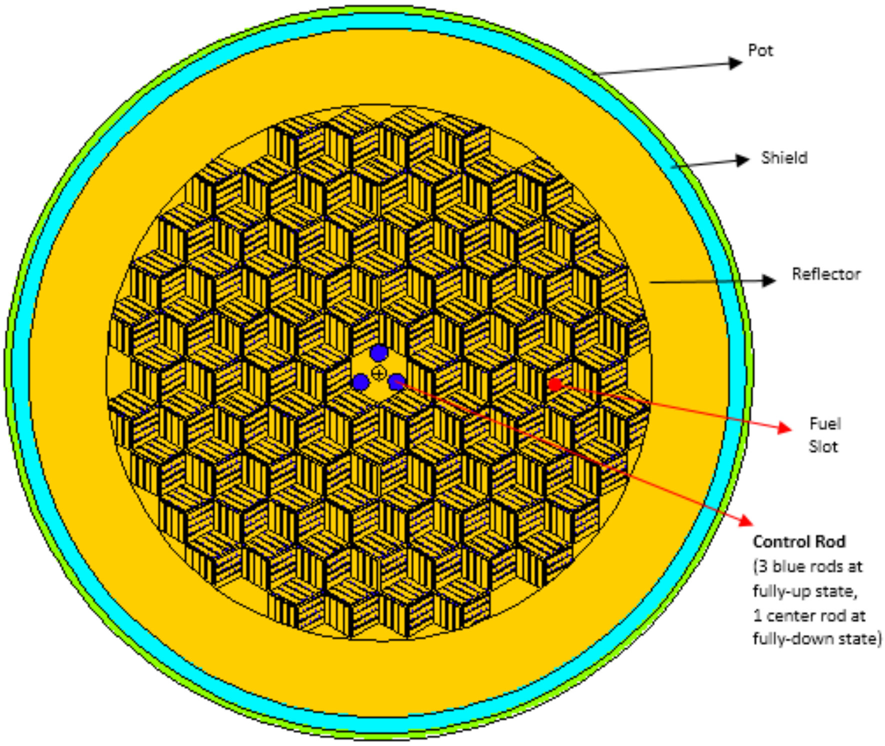


FIGURE 1. Core of TMSR-500

The core design of the TMSR-500 has 2 types of control rods, shutdown rods (which is blue) using gadolinium and regulating rods (the center one) using graphite. When the TMSR-500 operates the shutdown rod is in a fully-up state, while the regulating rod is in a fully-down state. The outermost layer pot or reactor core vessel uses SUS316H (316H Stainless steel) material, the next layer is a shield made from B_4C. The material used for moderators is graphite, as well as reflector materials. Liquid salt that functions as the main fuel as well as the coolant uses a mixture of NaF, BeF_2, ThF_4, and UF_4 [4].

The geometry that has been compiled on running using a PC with 200,000 neutron histories were tracked takes ± 20 hours for each run. The criticality of the TMSR-500 designed in this research was calculated by the MCNP6 program. The multiplication factor of neutrons or keff from the output produced shows the criticality of the reactor design that has been made. For the TMSR-500, the reactor design is declared in critical condition if the value is $1.00 < k_{eff} < 1.02$ [7].

In addition to calculating the keff value to determine the criticality of the design, validation of the geometry is also needed by analyzing the neutron energy spectrum on the TMSR-500. The neutron energy spectrum can be determined by calculating the value of the neutron flux in the fuel and moderator. To calculate the flux in cells from geometry, MCNP6 requires a tally flux card (F4) and an En tally card to obtain flux values in the specified energy range [8]. The range of neutron energy used is thermal neutron energy, epithermal neutron, and fast neutrons.

Bubbles or voids can occur due to the formation of fission products in gases which causes the reduced density of the fuel [5]. Therefore, this study uses variations in the void fraction with assumed variations in the reduction in fuel density. The variation in the reduction in fuel density used is 0%, 10%, 20%, 30%, and 40% with a void of 0% is the value of the fuel density during normal conditions.
The reactivity value is obtained from the value of k_{eff} with Equation 1.

$$\rho = \frac{\Delta k_{eff}}{k_{eff}} = \frac{k_{eff-1}}{k_{eff}} \qquad (1)$$

where k_{eff} is a neutron multiplication factor, the output of the MCNP6 program.

ISSN: 1412-3258 © 2020 BAPETEN – FMIPA UI

Then for k_{eff} of each void variation, the reactivity value was calculated. From the results of the void reactivity can be obtained the coefficient of void reactivity using Equation 2.

$$\alpha = \frac{\partial \rho}{\partial \phi} \quad (2)$$

where :
α = void reactivity coefficient
ρ = reactivity
ϕ = void fraction

RESULT AND ANALYSIS

Based on the results of running using MCNP6 at 977K, the value of k_{eff} was obtained $1.00 < k_{eff} < 1.02$. The k_{eff} value at 977K was obtained through the interpolation of the k_{eff} at 900K and 1200K, valued at 1.01804 ± 0.00008. **TABLE 1** displays the k_{eff} values from the simulations carried out at 900K and 1200K, and the k_{eff} obtained at the reactor operating temperature is 977 K.

TABLE 1. Values of k_{eff} at 977K from interpolation between 900K and 1200K

Temperature (K)	k_{eff}
900	1.02039±0.00008
977 (Interpolate)	1.01804±0.00008
1200	1.01124±0.00009

The results in **TABLE 1.** show that the simulation design of the TMSR-500 when operating at 977K was in a critical state.

Next is the evaluation of the energy spectrum as validation for the design geometry of the TMSR-500 that has been made. The neutron flux calculation can be done by the MCNP program by adding an F4 tally card (flux in cells). Cells whose neutron flux values are calculated are fuel cells and moderators, results can be seen in **TABLE 2** where neutron energy is only divided into thermal energy and fast energy.

TABLE 2. Neutron flux values of thermal and fast energy in fuel cells and moderators

Position	Neutron Flux ($n.cm^{-2}/s$)	
	Thermal Energy (10^{13})	Fast Energy (10^{12})
Fuel	2.0469 ± 0.0002	8.5209 ± 0.0002
Moderator (at fuel log)	2.5648 ± 0.0002	8.4983 ± 0.0002
Moderator (at rod log)	3.3651 ± 0.0014	8.9343 ± 0.0019

TABLE 3. k_{eff} values and reactivity change for each void fraction variation

Void (%)	k_{eff}	Reactivity Change (%Δk/k)
0	1.01804±0.00008	0
10	1.02641±0.00008	0.00801
20	1.03436±0.00007	0.01549
30	1.04140±0.00007	0.02203
40	1.04700±0.00007	0.02717

Based on **TABLE 2.** it can be seen that the value of thermal energy flux is greater than the value of fast energy flux, applicable to all three positions. This proves that the TMSR-500 is a thermal reactor, which means the reactor operates on thermal neutron energy. The moderator made from graphite has the role of slowing down the speed of neutron energy rapidly into thermal is also one of the factors why the TMSR-500 reactor is a thermal reactor [9]. A comparison of neutron flux values from the three positions is more clearly seen in **FIGURE 2**.

On the thermal reactor core, the maximum thermal neutron flux value is in the moderator and the maximum fast neutron flux value is in the fuel. As the edge of the neutron flux value decreases, it gets smaller, so that the highest thermal flux value is in the moderator cell (at rod log), where the log is in the center of the reactor core. The highest fast neutron flux value is in the fuel [10]. From the data in **TABLE 2**, it has been proven that it is in accordance with the literature.

Based on **FIGURE 3**, the void reactivity coefficient obtained from the data in **TABLE 3** using Equation (2) is positive (0.068 ± 0.003) %Δk/k/%void. The value of k_{eff} from each void fraction shows that when the void fraction increases, the reactivity value also increases. However, the TMSR-500 is an under-moderated reactor so the void reactivity is positive [6]. In previous studies [11] the MSR Fuji-12 reactor also had a positive void reactivity value.

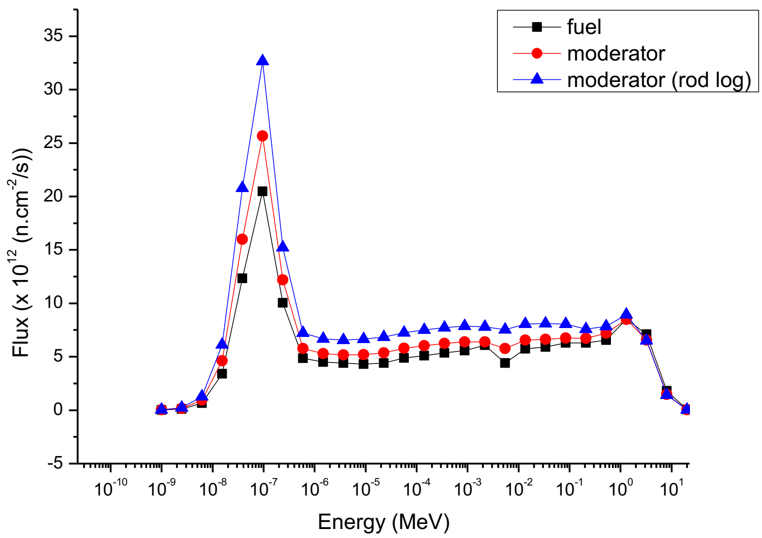


FIGURE 2. Graph of neutron energy spectrum in the fuel and moderator of the TMSR-500

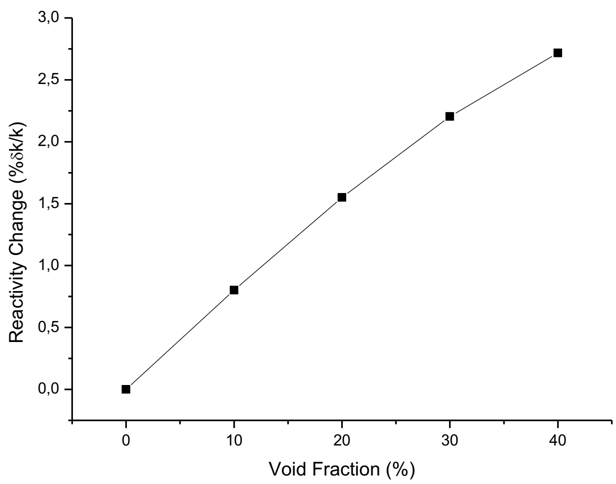


FIGURE 3. Graph of void fraction versus reactivity change

CONCLUSION

From the results of this research, it can be concluded that the design of the TMSR-500 operating at 977K is in a critical condition with a keff value of (1.01804 ± 0.00008). With a graphite based moderator whose role is to slow the rate of neutron energy rapidly into thermal, the neutron energy spectrum of the TMSR-500 is a thermal neutron spectrum. The coefficient of void reactivity obtained is positive value of (0.068 ± 0.003) %Δk/k /%void. Positive void reactivity is assumed to be caused by the condition of the TMSR-500 reactor which is under-moderated, then TMSR-500 declared safe. These results can be considered for further research analyzing other aspects of reactor safety, such as the coefficient of temperature reactivity and engineered safety (defense in depth).

REFERENCES

[1] W.A. Harto, "Prospek MSR Untuk Penggunaan Thorium di Indonesia", *Naskah Akademik* (Jurusan Tehnik Fisika UGM, 2016).

[2] J. Serp, M. Allibert, O. Benes, O. Delpech, O. Feynberg, D. Zhimin, Progress in Nuclear Energy, **77,** 308-319 (2014).

[3] M.S. Lumbanraja, E. Liun, Jurnal Pengembangan Energi Nuklir, **20(1),** 53-60 (2018).

[4] J. Devanney, *Thorcon The Do-able Molten Salt Reactor Design Control Document*. (Martiangle, Inc, 2015).

[5] D.B. Nugraha, A.W. Harto, Sihana, Jurnal Teknologi Reaktor Nuklir, **19(1),** 1-10 (2017).

[6] I. Singh, A. Gupta, U. Kannan, Nucl. Sci. Engin. **191(2),** 161-177 (2018).

[7] M. Fratoni, D. Shen, G. Ilas, J. Powers, *Molten Salt Reactor Experiment Benchmark Evaluation (Project 16-10240)*. (Nuclear Engineering University of California, Berkeley : United States. 2020).

[8] Suwoto., Zuhair. *Jurnal Sains dan Teknologi Nuklir Indonesia*. **17(2)**, 107-122 (2016).

[9] T. Griffiths, J. Tomlinson, R. O'Sullivan, *MSR Review – Feasibility of Developing a Pilot Scale Molten Salt Reactor in the UK*. Energy Process Developements Ltd, (2015).

[10] M. El-Wakil, *Nuclear Heat Transport*, (International Textbook Co, Michigan, 1971).

[11] N. Suzuki, Y. Shimazu, Jrnl. Nucl. Scie. Tech. **45(6),** 575-581 (2008).

Comparative Analysis of Neutron Fluence on Graphite Moderators between TMSR-500 and MSRE on Fast Neutron Energy (1 MeV - 10 MeV)

Dwi Margiyanto[1], Riyatun[1], Suharyana[1], Utari[1], Azizul Khakim[2]

[1]*Department of Physics, Faculty of Mathematics and Natural Science, Sebelas Maret University, Surakarta, Indonesia*
[2]*Nuclear Energy Regulatory Agency of Indonesia (BAPETEN)*
Jalan Gajah Mada No. 8 Jakarta Pusat

Correspounding author: suharyana61@staff.uns.ac.id

Abstract. Neutron flux and neutron fluence analysis on graphite moderator MSRE and TMSR-500 (ThorCon) was performed by simulation using MCNP6. Neutron flux distribution analysis and neutron spectrum using a kcode of 200,000 particles and 250 cycles to obtain normalized neutron flux values. Absolute neutron flux calculations were performed to obtain the neutron fluence value. The absolute neutron flux calculation on MSRE and ThorCon was carried out with the assumption of a power of 10 MWe with the operating temperature of each reactor, namely 920 K and 977 K. Neutron fluence calculations were carried out on the fast neutron energy spectrum with a range between 1 MeV - 10 MeV. The results of the calculation of the fast neutron flux in the graphite moderator MSRE and ThorCon, the highest energy is 1.35 MeV of 2.99×10^{13} n/cm^{2}s and 1.63×10^{12} n/cm^{2}s, respectively. The highest neutron fluence in the graphite moderator MSRE and ThorCon, respectively 3.76×10^{21} n/cm^2 and 2.06×10^{20} n/cm^2 for an operation time of 4 years. Damage to the graphite material can occur due to radiation caused by high energy neutrons (E> 0.1 MeV) when it reaches a neutron fluence of 4 x 10^{22} n/cm^2. Neutron fluence on the ThorCon graphite moderator showed no radiation damage because it was still much smaller than the maximum accepTABLE radiation limit. The results obtained also showed that the neutron fluence in the ThorCon graphite moderator was smaller than that of MSRE. These results prove that the ThorCon design is safe and ready to operate for 4 years.

Keywords: MSRE, ThorCon, neutrons fast fluence, neutrons flux

INTRODUCTION

World population growth is increasing over time. The world's population is expected to be more than 9 billion by 2050 [1]. The total population growth is directly proportional to the amount of energy demand, meaning that the higher the population growth, the higher the energy consumption or use. Indonesia's energy demand is estimated to reach 412 Million Tonnes of Oil Equivalent (MTOE) by 2025 [2].

Coal and petroleum are still the main sources of fossil energy in Indonesia. In addition, energy derived from fossil fuels can lead to climate change, acid rain, and combustion waste which can disrupt people's health. Therefore, Indonesia needs to replace primary energy sources that come from fossil fuels to energy sources that are cleaner and more efficient. One of the energy sources that can be used is nuclear energy. Nuclear energy can produce energy with a large capacity, is efficient and environmentally friendly.

Nuclear energy also has several obstacles. Current common power reactors, for example, the Pressurized Water Reactor (PWR) type use the main fuel, namely uranium (235U). 235U is the fissile material used as fuel in power reactors, while the 235U content in nature is only about 0.7%. According to the Nuclear Energy Agency (NEA), uranium resources amount to 5.5 million tonnes and 10.5 million tonnes which have not been discovered. A viable uranium supply for the current reactor will last approximately 80 years at current consumption rates [3]. Therefore, it is necessary to find other nuclear materials as a substitute for uranium and more advanced reactor design to be developed. One alternative nuclear material that can be used as nuclear fuel for advanced reactor designs is thorium (Th) for the Molten Salt Reactor (MSR) type reactor.

The Molten Salt Reactor is a generation IV reactor that was first developed by the Oak Ridge National Laboratory (ORNL). The MSR reactor can operate at high temperatures (704 ºC) at ordinary pressure (1 atm) with an efficiency of 45% [4]. The advantage of MSR compared to current reactor types is a simpler core design by using molten salt as fuel as well as coolant. Other advantages of MSR are low reactivity, high burn-up, low waste production, and high capacity [5].

Molten Salt Reactor Experiment (MSRE) is one of the One Fluid-MSR designs developed by ORNL. The main features of MSRE are [6]:

- MSRE is a single liquid fuel reactor concept where the fuel flows through a graphite line.

- MSRE is a cylinder reactor design with a power of 10 MWe.
- Another component that comes into direct contact with the salt fuel is made of Hastelloy nickel (INOR-8).
- In the center of the core there are three control rods and a graphite assembly.

MSRE is an experimental liquid salt reactor that experienced criticality in 1965 and operated until 1969. The data obtained during the MSRE operation were significant. These data are used as a yardstick to verify the MSR model currently being developed [7]. One of the MSRs currently being developed and planned to be built in Indonesia is the TMSR-500 (ThorCon). MSR ThorCon is a reactor that can produce a total power of 1,000 MWe. The total power is generated from 4 modules, where each module generates a power of 250 MWe [8]. MSRE is a reactor that operates for 13,000 hours, while MSR ThorCon operates for 4 years. Before the ThorCon reactor is built in Indonesia, an analysis is needed to prove that the ThorCon MSR is ready to operate. One of the parameters that can be used to test ThorCon MSR is the analysis of the neutron fluence.

Neutron fluence or what is known as the neutron dose is a time integral of the neutron flux density. Neutron fluence is expressed as the number of neutrons per cm2. Neutron fluence is also used as a measure of the burn-up of fuel in reactors. Because the rate of combustion is proportional to the neutron flux. The burn up over a while is proportional to the neutron flux and time ($F = \Phi.t$ or $F = \int \Phi \, dt$) is known as the neutron fluence. The amount of a neutron fluence is the neutrons per cm2, but is often expressed in terms of neutrons per kilobarn [9].

Neutron fluence is an important parameter in technology and safety in reactor operation. Neutron fluence is used to analyze radiation damage to the material over a while. In other words, neutron fluence analysis is used to test the resistance of the reactor material over a certain period. Analysis of the neutron fluence can be carried out on a graphite moderator in the reactor core. The analysis was carried out on the graphite moderator because it is one of the main components and plays an important role in the reactor. Neutron fluence analysis data on ThorCon were compared with MSRE to verify the data. MSRE is a type of liquid salt reactor which is almost the same as MSR.

Damage to the reactor core material can be caused by radiation exposure. Damage to the material in the reactor due to radiation exposure results in shorter service life. The higher the exposure a material receives, the shorter its useful life. Damage to the material in the reactor due to radiation will occur if it exceeds the material's neutron fluence limit. Therefore, an analysis of the neutron flux distribution and the neutron spectrum was carried out to determine the neutron fluence value in the reactor material. The resistance of a material in the reactor can be determined by analyzing the neutron fluence. Neutron fluence calculations were carried out on the graphite moderator TMSR-500 and MSRE reactors.

Monte Carlo is a probabilistic or stochastic analysis method with microscopic calculations to obtain macroscopic system behavior. Monte Carlo uses a number of particles to be sampled to represent the system as a whole [4]. Monte Carlo N-Particle (MCNP) is a computer code to probabilistically simulate neutrons, photons, and electrons when they undergo fission reactions and stochastic interactions with matter. MCNP6 is a combination of features from MCNP5 and MCNPX. The MCNP code is usually used for the needs of a fairly large nuclear area. For example, MCNP uses are reactor design, nuclear criticality safety, medical physics, nuclear safeguards, and others.

This study aims to analyze the calculation and comparison of neutron fluence in graphite moderator between ThorCon and MSRE using the Monte Carlo method with the MCNP6 program. If the neutron fluence moderator of graphite on the ThorCon MSR is smaller than the MSRE, then the ThorCon design can be said to be ready to operate in terms of fluency.

METHODOLOGY

1. MSRE Core Model

The simulated MSRE reactor core geometry refers to the ONRL document data. The material composition and geometry of MSRE are shown in **TABLE 1** and **TABLE 2**.

TABLE 1. Geometry and parameters of MSRE core [10].

Parameters	Value
Active cylinder	Height: 166.4 cm Diameter: 142 cm
Side length of element	5.08 cm
Height of upper/lower plenum	20 cm
Thickness of annular plenum	2.54 cm
Thickness of vessel	1.42 cm

TABLE 2. Materials on MSRE core [10].

Parameters	Value
Fuel salt	65% LiF: 29.1% BeF_2: 5.0% ZrF_4: 0.9% UF_4 Density: 2.27 g/cm^3
Graphite	Density: 1.86 g/cm^3
Nickel base alloy	Density: 8.86 g/cm^3
Control rod	Propotion: 70% Gd_2O_3: 30% Al_2O_3 Density: 5,87 g/cm^3

2. ThorCon Core Model

The simulated reactor core geometry refers to the ThorCon MSR design. The material composition and geometry of ThorCon are shown in **TABLE 3** and **TABLE 4.**

TABLE 3. ThorCon core size specifications [8].

Parameters	Value
Pot Diameter (cm)	518,1
Pot Height (cm)	571,1
Core diameter (cm)	343
Core Height (cm)	378
The number of fuel salt logs	84
Log apothem (cm)	19,055
Slab thickness (cm)	4
Slot thickness (cm)	0,6
Plenum height (cm)	50

TABLE 4. Materials in ThorCon core [8].

Parameters	Value
Fuel salt	76% NaF: 12% BeF_2: 9,8% ThF_4: 2,2% UF_4 Density: 2,95 g/cm^3
Graphite	Density: 2,66 g/cm^3
Plenum	Proportion: 99% Graphite: 1% Fuel salt Density: 2,66 g/cm^3

3. Neutron Flux Distribution Analysis and Neutron Spectrum

Neutrons (E <0. Neutron flux distribution analysis and neutron spectrum at ThorCon and MSRE reactors were carried out on graphite moderator in the reactor core. Neutron flux distribution analysis was carried out in 3 energy groups, namely thermal 625 eV), intermediate neutrons (0.625 eV <E <100 keV), and fast neutrons (E> 0.1 MeV). The reactor power is simulated at 10 MWe. Neutron flux distribution analysis was performed using tally card F4 in kcode. The results obtained from the MCNP are still normalized flux values. To get an absolute flux value a conversion factor is required. The amount of absolute flux [11]:

$$S = \frac{p\underline{v}}{(1.6022\times10^{-13})w_f k_{eff}} \tag{1}$$

p = reactor thermal power (Watt)
$\underline{v}$ = the average number of neutrons released per fission (neutron/fission)
w_f = the effective energy released per fission (MeV/fission)
k_{eff} = the effective neutron multiplication factor

$$\Phi = S.\Phi_{F4} \tag{2}$$

Φ = the actual total neutron flux (neutron/cm^2 s)
Φ_{F4} = MCNP calculation output results (1/cm^2)

4. Neutron Fluence Analysis

Neutron fluence is the product of the neutron flux and the duration of the reactor operating time. The maximum neutron fluence that graphite can accept in the fourth generation reactor is 4 x 10^{22} n/cm^2 (E> 0.1 MeV) [12]. Neutron fluence analysis was performed at the reactor operating time for 13,000 hours and 4 years. The magnitude of neutron fluence [13]:

$$F = \Phi . t \text{ or } F = \int \Phi \, dt \quad (3)$$

with:
Φ = neutron flux (neutron/cm^{2}s)
t = reactor operating time (s)

RESULTS AND DISCUSSION

1. MSRE and ThorCon core models

MSRE and ThorCon reactor vessel and core models have been made based on the material composition and geometry data shown in **FIGURE 1(a)** and **FIGURE 1(b).** At Thorcon, the reactor core is surrounded by a reflector made of graphite. Whereas in MSRE, the reactor core is surrounded by a plenum which is a homogeneous salt fuel.

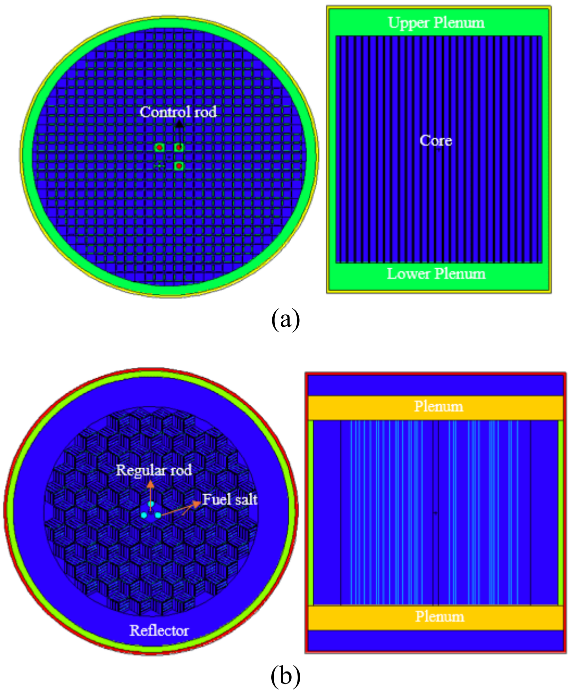


FIGURE 1. MCNP core model (a) MSRE and (b) ThorCon.

The results of the criticality calculation on MSRE and ThorCon yield k_{eff} values of 1.00311 ± 0.00014 and 1.01804 ± 0.00008, respectively. The k_{eff} value is obtained using the MCNP6 program with the operating temperature of each reactor, namely 920 K and 977 K. The k_{eff} value of the ThorCon reactor is calculated by assuming that all the shutdown control rods are pulled out of the reactor core (fully-up). The k_{eff} results obtained can be trusted because both reactors are in critical condition. This is in accordance with research conducted by [14] with the k_{eff} value on MSRE of 1.0598. For the ThorCon reactor, the k_{eff} value is close to the result obtained by [15] of 0.99818 ± 0.0003.

2. Neutron Flux Distribution and Neutron Spectrum

Neutron flux distribution analysis and neutron spectrum at ThorCon and MSRE reactors were performed using MCNP6 based on the geometry and models that have been made. Neutron flux distribution analysis and neutron spectrum were carried out on a graphite moderator using 200,000 particles and 250 cycles of code. The distribution of the neutron spectrum is carried out over an energy range of 10^{-8} MeV to 10 MeV which is used to obtain normalized neutron flux. To determine the absolute flux of neutrons produced in a graphite moderator, equations

(1) and (2) are used. Neutron flux distribution calculations on MSRE and ThorCon were performed using a power of 10 MWe.

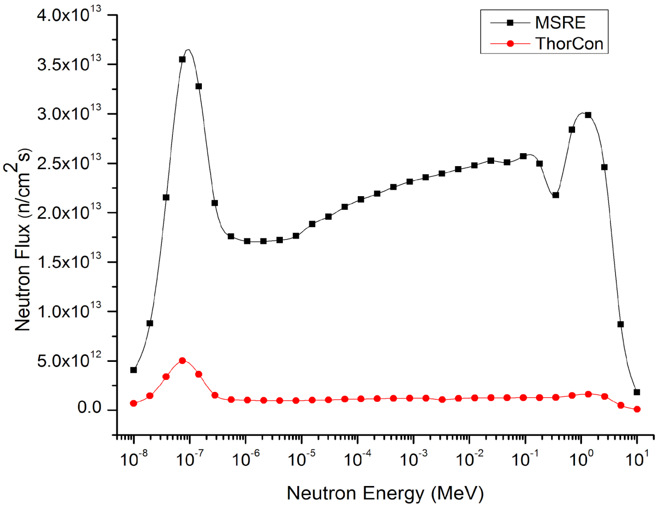


FIGURE 2. Distribution of neutron fluxes with neutron energies of 10^{-8} MeV - 10 MeV in a graphite moderator.

The results of the neutron spectrum calculations for the MSRE and ThorCon graphite moderators are plotted in **FIGURE 2**. **FIGURE 2** shows the distribution of fluxes with neutron energies of 10^{-8} MeV - 10 MeV in moderators of ThorCon and MSRE graphites. The graph shows that there is a higher peak in thermal neutron energy ($<6.25 \times 10^{-7}$ MeV) which indicates that both reactors are operating on the thermal neutron spectrum. The decrease in fast neutrons in the moderator is more than the decrease in the thermal neutron flux indicates that the moderation process and the absorption of fast neutrons are more in the passing medium. This shows the characteristics of the reactor core using graphite as a moderator.

The density of the ThorCon graphite moderator is greater than that of the MSRE graphite moderator. The moderator density value for ThorCon graphite is 2.66 g/cm^3, while at MSRE it is 1.86 g/cm^3. The neutron flux value in the MSRE graphite moderator was greater than that of ThorCon. That is, the graphite moderator on ThorCon has a greater cross-sectional view than MSRE. This causes the neutron energy absorbed by the TMSR-500 moderator to be greater than MSRE when passing through the medium in its path. The fast neutron flux values in the graphite moderator are shown in **TABLE 5.**

TABLE 5. Fast neutron fluxes in graphite moderator MSRE and ThorCon.

Energy (MeV)	Φ (n/cm^{2}s)	
	MSRE	ThorCon
1.35	2.99×10^{13}	1.63×10^{12}
2.63	2.46×10^{13}	1.39×10^{12}
5.12	8.70×10^{12}	5.00×10^{11}
10.00	1.81×10^{12}	1.06×10^{11}

3. Neutron Fluence

The results of the neutron fluence calculation on the graphite moderator were carried out at the reactor operating time for 13,000 hours and 4 years. The calculations were carried out during this time because the MSRE operating time was 13,000 hours, while the ThorCon operating time was 4 years. The neutron fluence values for fast neutron energy in graphite moderators are shown in **TABLE 6.**

TABLE 6. Neutron fluence in graphite moderator MSRE and ThorCon.

Energiy (MeV)	Neutron Fluence (n/cm^2)			
	13000 hours		4 years	
	MSRE	ThorCon	MSRE	ThorCon
1.35	1.40×10^{21}	7.64×10^{19}	3.76×10^{21}	2.06×10^{20}
2.63	1.15×10^{21}	6.49×10^{19}	3.10×10^{21}	1.75×10^{20}
5.12	4.07×10^{20}	2.34×10^{19}	1.10×10^{21}	6.31×10^{19}
10.00	8.48×10^{19}	4.97×10^{18}	2.29×10^{20}	1.34×10^{19}

The ThorCon design still doesn't have a license as the reactor is still in the conceptual design. A commercial ThorCon reactor that can be implemented has to be tested [16]. One of the parameters that can be used to test the readiness of the ThorCon design is the ratio of the neutron fluence to the MSRE graphite moderator. The MSRE was a successful experimental reactor on which the knowledge ThorCon was built on. **TABLE 6** shows that the neutron fluence value in the graphite moderator TMSR-500 are much smaller than that of MSRE for both 13,000 hours and 4 years of operating time. The result of the neutron fluence calculation can be said that the ThorCon MSR design fulfills one of the requirements to be ready for operation.

The duration of the reactor operation affects the neutron fluence value in the graphite moderator. The longer the reactor operating time, the more the neutron fluence value increases. Increased neutron fluence results in embrittlement and a decrease in the quality of graphite due to radiation destruction. The life span of using graphite as a moderator is very dependent on the power density and reactor operating time. The length of the reactor operating time affects the durability and quality of the graphite moderator. High exposure to graphite moderators can cause damage to its internal structure.

The ThorCon reactor is a reactor designed with 4 modules that produce a total power of 1,000 MWe, where each module produces 250 MWe of power with graphite as the moderator. The fast neutron fluence calculation on the ThorCon graphite moderator was carried out at full power of 250 MWe. The fast fluence neutron value at an energy of 1.35 MeV is 0.51 × 1022 n/cm2 for a 4-year operating time. The neutron fluence on the ThorCon graphite moderator obtained at high neutron energies (> 0.1 MeV) is still much smaller than the accepTABLE maximum limit for graphite, which is 4×10^{22} n/cm^2 [12]. Thus, from the side of the fast fluence neutron in the graphite moderator, it can be said that the ThorCon reactor is still safe to operate for up to 4 years.

CONCLUSION

The fast neutron fluence with an energy of 1 MeV to 10 MeV in the ThorCon graphite moderator is much smaller than that of MSRE during the operation time of 13,000 hours or 4 years. The highest fast neutron fluence was at an energy of 1.35 MeV for 4 years at MSRE and ThorCon, respectively 3.76 × 1021 n/cm2 and 2.06 × 1020 n/cm2.

Neutron fluence on ThorCon graphite moderator fast neutron energy spectrum at full power of 250 MWe, namely 0.51 × 1022 n/cm2 for 4 years operation time. The results obtained are also far from the threshold for radiation damage to graphite (4×10^{22} n/cm^2). With the neutron fluence study, it can be concluded that the ThorCon reactor design is still safe and can be said to be ready to operate with an operating time of 4 years.

REFERENCES

[1] Bahman, Z., & Patrick, M. *Thermodynamics in Nuclear Power Plant Systems*. (Springer International Publishing, Switzerland, 2015).

[2] BP Statistical Review of Word Energy. Databoks (2017), available at https://databoks.katadata.co.id /datapublish/2017/10/14/20-tahun-konsumsi-energi-indonesia-meningkat-2-kali-lipat.

[3] Fetter, Steve. Scientific American (2009), available at https://www.scientificamerican.com/article/how-long-will-global-uranium-deposits-last/.

[4] Bielajew, A.F. *Fundamental of the Monte Carlo Method for the Neutral and Charged Particle Transport*. (The University of Michigan, Michigan, 2001).

[5] Ashraf, O., Smirnov, A.D., & Tikhomirov G.V. Jrnl Phys: Conf. Ser. 1189 012007 (2019).

[6] Robertson, R.C. Tech Rep ORNL-TM-728. Oak Ridge National Lab. (1965).

[7] Haubenreich, P.N., & Engel, J.R. Nucl. Applic. & Tech. **8** (1970).

[8] Devanney, J. ThorCon (2015), available at http://thorconpower.com/docs /exec_summary2.pdf.

[9] Fei, T., Feng, B., & Heidet, F. Ann. Nucl. Erg. (2019).

[10] Zhu, G.F., Yan, R., Peng, H.H., Ji, R.M, Yu, S.H., Liu, Y.F., Tian, J., & Xu, B. Nucl. Sci. Tech. **30**-**34** (2019).

[11] Dawahra, S., & Khattab, K. Ann. Nucl. Erg. **38** (2011).

[12] Campbell, A.A., Katoh, Y., Snead, M.A., & Takizawa, K. Oak Ridge, TN 37831, ISSN 0008-6223 (2016).

[13] Lamarsh, J.R. *Introduction to Nuclear Reactor Theory, 2nd ed*. (Addison-Wesley, Boston. 1983).

[14] Kophazi, J., Sziebert, M., Feher, S., & Leege, F.A JAERI-Conf. 019 (2003).

[15] Fei, T., Feng, B., & Heidet, F. Ann Nucl. Erg. (2019).

[16] Lumbanraja, S.M., & Liun, E. Jurnal Pengembangan Energi Nuklir **20**(1) (2018).

Preliminary Study on Radiation Protection and Safety for Itinerant Workers

Slamet Wiyuniati[1], Sutrasno[2]

[1]*Education and Training Center of BATAN*
[2]*Center for Isotope and Radiation Applications of BATAN*

E-mail: [1]swiyuni@batan.go.id, [2]sutrasno@batan.go.id

Abstract: In more than a decade, the government of Indonesia has been regulating radiation protection and safety of the utilization of ionizing radiation sources for protecting people, workers, and the environment. The government of Indonesia has been implementing this regulation on various nuclear facilities to act as required guidance for employers, managers, radiation protection specialists, and radiation workers to control one's occupational exposure to radiation. In terms of radiation workers, the content of the regulation itself only focusing on radiation protection and safety for workers of a particular employer or organization, and it does not pay enough attention to workers who were employed to work in more than one facility for a certain period of time. In Indonesia, the number of nuclear facilities is only getting higher, and in return, the possibility of increasing numbers of itinerant workers per year is only plausible. The itinerant workers are "mobile" workers who work for more than one nuclear facility at several different times with varying durations. The task of those itinerant workers is to ensure the safety of people, other employees, and the environment from the harmful amount of radiation exposure, and thus often required highly trained personnel with plenty of experience. Effective radiation protection and safety regulation is an essential component in order to control radiation exposures in every nuclear facility. Unfortunately, at this moment in time, there is not enough technical guidance on radiation protection and safety to follow by nuclear facilities and their workers in Indonesia. Therefore, the government of Indonesia should take this situation seriously and establish a proper technical regulation for every nuclear institution in Indonesia; otherwise, this undertaking venture would become a much more serious problem in the future. This paper describes the technical and legal aspects of radiation protection and safety for itinerant workers who work at more than one nuclear facility.

Keywords: Itinerant workers, radiation protection and safety, Nuclear Facility, Principal Employer, Contractor

INTRODUCTION

Itinerant worker is a high mobility worker who works at more than one facility and often relocates from one facility to another. Employers also referred itinerant workers as contract workers, seasonal workers, migrant workers, and temporary workers. The role of those itinerant workers varies from maintenance staff, quality assurance personnel, non-destructive inspection workers, training instructors, medical staff, security staff, contractor workers, etc. [1]

The main parties involved in the completion of certain projects in an establishment that employs itinerant workers are the principal employer, contractor, and the itinerant workers themselves. In general, the hiring process of these itinerant workers starts when the principal employer requests a service of a particular contractor to provide workers in order to meet the workforce requirement of a certain project, and the workers supplied by the contractor would send daily or weekly reports to the contractors as their employer. The two main setbacks of this process are the difficulty directing and monitoring the works carried out by these itinerant workers by the principal employer.

Itinerant workers could also be categorized into two different categories depending on their employment territory. Domestic itinerant workers are workers who work in several installations within one country; meanwhile, international itinerant workers are workers who operate in several installations across various countries.

The International Atomic Energy Agency (IAEA) defines itinerant workers as occupationally exposed workers who work in supervised and/or controlled areas at one or more locations, and are not employees of the management of the facility where these said workers are operating. [2, 3]. For example, Figure 1 shows the numbers of itinerant workers hired by BATAN's education and training center who were also employed in other domestic nuclear facilities across the years.

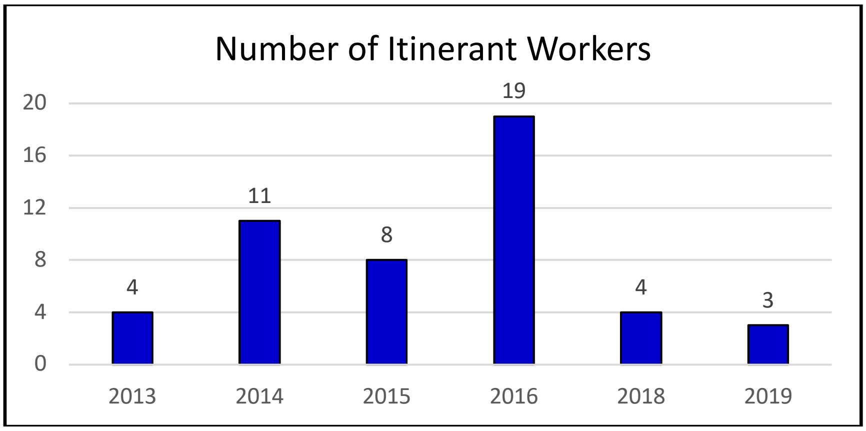


FIGURE 1. BATAN's Education and Training Center Itinerant Workers Data from 2013 to 2019.

The contribution of the itinerant workers to the workforce in the Indonesian nuclear industry is increasing significantly due to the numerous installations of new nuclear facilities. As shown in **FIGURE 2**, the number of Indonesian nuclear facilities consisted of Medical and Industrial sectors reached 3807 facilities as of September 2020. [4]

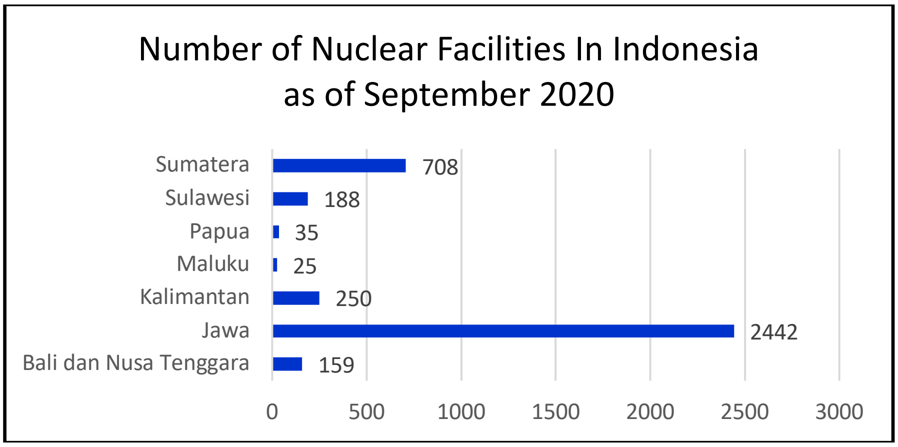


FIGURE 2. Number of Nuclear Facilities in Indonesia as of September 2020.

In the nuclear industry, itinerant workers are employees who were assigned by their contractors or employers with the risk of direct exposure to radiation in a nuclear facility or installation. They may work in various supervised and often controlled areas of nuclear facilities, but the collective dose of these controlled radiation exposures from several facilities could affect the protection and safety of those itinerant workers. The matter in hand will become more severe for itinerant workers with a wider range of work. In some other cases, inexperienced contractor's itinerant workers when working on a project requiring a radioactive source decided to bring their own radioactive source provided by their contractor to a certain facility without a proper procedure and regulation. Complete mishandling attempt by the itinerant workers with insufficient knowledge to minimize the exposure of a radioactive source could be dangerous to other workers of a nuclear facility and the itinerant workers themselves.

The government's regulation on radiation protection and safety of the utilization of ionizing radiation does not clearly state the problems mentioned above. The available regulation does not cover enough guidance for all the parties involved to handle these issues.

This paper describes the legal and technical aspects of radiation protection and safety for itinerant workers, especially those assigned by several nuclear facilities in Indonesia for further safety regulation reviews.

METHODOLOGY

This proposal is made based on a preliminary study on some related documents of radiation protection and safety, and working experiences of delivering lectures on radiation protection courses since the year of 2008.

RESULT AND DISCUSSION

The IAEA General Safety Guide Number 7 of the year 1999 on Occupational Radiation Protection provides basic guidance on the aspects of radiation protection for the itinerant workers to support the needed cooperation between employers of the workers (Principal Employer and Contractor). The purpose of this cooperation is to avoid overlapping responsibilities between Employers over itinerant workers' management. [1] Additionally, the General Safety Requirement Part 3 number 23 of the IAEA Basic Safety Standard of the year 2014 provides limited guidance to nuclear facilities regarding the procedure of employing itinerant workers to carry out tasks with direct exposure to radiation. It also provides a more detailed explanation regarding the cooperation between Principal Employer and Contractor [2]. The IAEA Safety Report Series Number 84 of the year 2015 suggests regarding the management of itinerant workers' activity in respect of their radiation protection and safety. It also recommends the maintenance of itinerant workers' regulated exposure to nuclear radiation [3].

In Indonesia, the Government Regulation number 33 of the year 2007 on Radiation Protection and Safety of the Utilization of Ionizing Radiation states that radiation safety must ensure workers' safety, people's safety, and the environment's safety from the dangers of radiation exposure. Every individual or organization that would take advantage of nuclear energy must have a license issued by a governmental regulatory organization after meeting all the necessary radiation safety requirements. It also mentions that the licensee could be a Principal Employer or an Independent Contractor [5]. The decree of BAPETEN Chairman Number 4 of the year 2013 states that the technical responsibility of workers' radiation safety falls under the licensee and the individual in charge of the radiation use. The person in charge itself could be a radiation worker, a radiation protection officer, or an individual pointed by the licensee. In terms of the legal aspect, the licensee will take full responsibility for the pointed individual [6].

There are various important aspects to ensure the radiation safety of itinerant workers and are requiring further reviews to avoid any possible legal issues arising from the additional workforce of a nuclear facility. The aspects in questions are the mutual agreement document between the principal employer and contractor, detailed tasks, dose constraint value, dose limit value, personal dose monitoring, medical check-up, possible required training, etc. The Principal Employer and the Contractor should be responsible for the success of a certain project, as well as the safety of those itinerant workers hired for the said project.

Agreement of Cooperation between Parties

A collaboration agreement is a legally binding agreement between parties prior of working on a certain project. This certain agreement should be based on the requirements of the Indonesia Civil Code articles No. 1320. An employment process of itinerant workers to be hired in more than one nuclear facility should be based on a legal agreement between the principal employer (as the first party) and the contractor (as the second party). This agreement is made by both parties to further clarify the responsibilities held by each party for the agreed project. The detailed content of this agreement should discuss the joint responsibilities to the necessary extent in order to comply with all the basic requirements of protection and safety for all the workers in the facility. This process of cooperation should cover some points of interest, such as the rights and obligations of both parties during the project, the number of radiation workers needed by the principal employer or the contractor, the work duration needed to complete the project, and the licensee.

In this agreement, contractors should ensure that their employees are suitably qualified for the project and should submit details of each employee's qualifications to the principal employer prior to commencing a project at a nuclear facility. Itinerant workers should not be allowed to work without the required training and certification in radiation protection. The principal employer should ensure that contractors carrying out work at a nuclear facility are using personnel who are competent. Accordingly, the competence of contractor personnel may need to be formally assessed and documented. Under certain circumstances, the principal employer may wish to specify site-specific competence requirements to be fulfilled before the contractor is permitted to do any on-site work.

One of the most important points from the agreement is deciding the man in charge of the radiation source at the facility. The one responsible for the radiation source should be the party with the licensee in order to properly control the exposure of the radiation source. The licensee should obtain all the necessary information of the itinerant workers, provides appropriate information to the employer or contractor, and provides both the workers and the employers with the relevant exposure records. In many cases, the contractor and its employees have little to no experience in dealing with radiation safety and protection, hence the lack of knowledge in providing the proper regulation for their radiation workers. In this case, it is the responsibility of the principal employer to apply the same level of protection and safety to the itinerant workers as its own employee.

The agreement should include a few cases of scenarios on the use of the radiation source since the source can be under the control of either the principal employer or the contractor. The project activities from the beginning to the end should be carried out on a scenario basis. A more complicated agreement scenario should be considered when both parties use their own radiation sources at the facility.

Period of Working Assignment

Itinerant workers are known to work at two different facilities. Both parties involved in employing these itinerant workers should monitor the dose of radiation exposure and regulate their working period to a certain duration instead of the continuous working duration. The working period of each worker at the nuclear facility should be elaborated in the agreement based on the radiation protection program made by both parties, as well as the National Law regarding labor. The working period must ensure that the dose constraint for itinerant workers is not exceeding the regulated limit.

Radiation Protection Programme

A radiation protection program for radiological evaluation should be a collaborative effort between the principal employer and the contractor. For a facility that uses radiation sources as part of its normal operation, the principal employer should carry out a prior radiological evaluation and safety assessment for its operations. Similarly, where the contractor has its own sources of radiation, it should carry out appropriate radiological evaluation and safety assessments for most of the nuclear facilities at which those sources are likely to be used. In a more complex situation, it may be possible to have a combination of both scenarios.

To ensure the safety and health of itinerant workers, a dose constraint and dose limit value for the itinerant workers should follow the provision of a regulatory organization. The amount of radiation in itinerant workers could be more stringent compared to those of normal radiation workers. The current dose limit of 15 mSv per year can be used as an initial dose constraint, and the effective dose limit could be 20 mSv per year. For example, the licensee can determine a dose limit value of the itinerant workers over the age of 18 years to be about 75% compared to the dose limit value of regular radiation workers [6]. Similarly, the licensee should also be able to determine an annual effective dose value of the itinerant workers; an authorized body or regulatory organization will investigate any exposure exceeding the regulatory rule.

The Activities of Itinerant Workers

The activities of the itinerant workers at a nuclear facility are systematically divided into stages of planning, executing, evaluating, and reporting. Before executing the work, there are a number of things to be considered at the planning step. The principal employer is needed to provide a nature of planned work to the itinerant workers, such as type of work, duration time, the target of work, operating procedures, accompanying officers, materials, facilities and equipment, the provisions of radiation protection and safety, the potential radiation and/or contamination hazards, personnel monitors and radiation protection equipment, other personnel protective equipment, etc.

At the workplace, the itinerant workers are obliged to follow the radiation protection and safety provisions, as well as the procedures to prevent an excessive dose of radiation exposure. Considering the development of potential hazards from radiation exposure and radiation contamination of those itinerant workers, a scheduled dosage check by the facility safety supervisor or radiation protection officer is required. Furthermore, the necessary protection and radiation safety measures should be appropriately taken to ensure the task run by the workers is still within the limits of health and safety regulations.

In case of a disruption of the equipment or the facility, the radiation worker should immediately report the problem to the facility manager or radiation protection officer. Before proceeding to repair the equipment or the facility in question, the facility safety supervisor or radiation protection officer must assess the potential hazards to prevent an excessive amount of radiation exposure to the workers. Moreover, in the case of any conditions that may lead to an emergency, the itinerant worker should also immediately report the situation to the facility safety supervisor or radiation protection officer. Following the report, the supervisor or radiation protection officer should take all the necessary action in accordance with the provisions of radiation protection and safety.

The itinerant workers will have to make a report of completion of work per the quality assurance system. The report must be approved by the supervisor and the facility manager. The report must contain at least the date of work, time of work, workplace or location, materials, equipment and facilities, job description, working target and result, obstacles of executing the assignment, and any other specific notes. Itinerant workers' job report is part of the report developed by the facility manager. This report should include radiation safety data, such as radiation exposure rate, surface contamination, and air contamination (if any) before, during, and after the project. For approval, the report then should be endorsed by the plant quality assurance manager. The report is then formally submitted by the contractor to the principal employer. This report may be used to evaluate the implementation of cooperation between parties.

Records of Occupational Exposure and Medical Check-Up

Due to its nature of work, the itinerant workers might work at several facilities and at any different times and for varying durations. In consequence, they might have increased doses of radiation from the multiple facilities within a certain period. These workers' doses should therefore be tracked over a long period of time. The responsibilities and arrangements for achieving this should be established and documented in physical and electronic forms by the principal employer and/or contractor. This personal dose of itinerant workers then should be informed to the itinerant workers. It is the responsibility of the employer of the itinerant workers to ensure that the worker's record of occupational radiation exposure is kept up to date.

The contractor should be responsible for the medical record of itinerant workers during a project. These medical records include a medical examination, counseling, and health management as a result of an excessive dosage of radiation. Itinerant workers should follow a medical test before the assignment, during the assignment, and after the termination of the assignment. The types of medical checks should be both a general and a special health test. The medical check-up result then should be documented in accordance with the quality assurance system and provision of radiation protection and safety. This medical check-up result should also be informed to the itinerant workers.

Case of Excessive Exposure

To avoid an excessive radiation dosage on the itinerant workers, the parties involved should make an effort to ensure that the received dosage is as low as possible in consideration of the social and the economic factors of itinerant workers. In the case of excessive radiation exposure, the contractor should be able to provide a medical follow-ups and bioassay samples of the workers when needed for further laboratory check-ups. Besides that, the contractor should also be responsible for itinerant workers' medical observation. This medical information of itinerant workers will be included in the report.

CONCLUSION

The contribution of the itinerant workers in the Indonesian nuclear industry workforce is increasing significantly due to the rapid growth of nuclear facilities in the country. Itinerant workers can be any type of workers who work at several nuclear facilities with a higher probability to be exposed to a source of radiation. Some radiation and radiological safety issues may arise from the employment of these itinerant workers.

In the aspects of radiation protection for the itinerant workers, the guidance issued by the IAEA should be followed and obliged by the principal employer, contractor, and the workers themselves. It is highly suggested that the employment of itinerant workers should be based on legal cooperation agreed by the principal employer and the contractor. The collaboration agreement itself has to follow all the requirements of the Indonesia Civil Code articles No. 1320.

The government of Indonesia has published a regulation on radiation protection and safety of the utilization of ionizing radiation sources to ensure the safety of people, workers, and the environment. In addition to that, a regulation that covers the radiation protection and safety for itinerant workers are in need to be legally and technically formulated based on the available regulations and the necessary safety procedure among nuclear facilities. In order to prevent arising issues over the safety and protection of itinerant workers, it is of utmost importance for the Indonesian Government to provide a safe working environment that could reliably control the health and safety of the itinerant workers. The regulatory organization should consider all the important variables such as mutual agreement document between the principal employer and contractor, the details of work, dose constraint value of itinerant workers, dose limit value of itinerant workers, records of occupational exposure, personal dose monitoring, medical check-up, and proper training, to ensure the safety and protection of itinerant workers, and provide proper technical guidance for every nuclear facility in Indonesia.

ACKNOWLEDGEMENT

We would like to express our sincere gratitude to all my colleagues at the education and training center of BATAN who has given us a chance to complete this paper. Many thanks also go to the director of education and training center of BATAN for his valuable support.

REFERENCES

[1] The IAEA General Safety Guide Number 7 (GSG-7) year 1999 on Occupational Radiation Protection (1999)
[2] The IAEA Safety Standard Series No GSR Part 3 the year 2014 on Radiation Protection and Safety of Radiation Source (2014)
[3] The IAEA Safety Report Series number 84 the year 2015 on Radiation Protection of Itinerant Worker (2015)
[4] https:www.balis.bapeten.go.id/portal/web/index.php/sites/ktun-per-provinsi
[5] The Ministry of State Secretariat of the Government of Indonesia, The Government Regulation Number 33 year 2007 on Safety of Ionizing Radiation and Security of Radioactive Sources (2007)
[6] The BAPETEN's Chairman Regulation Number 04 the year 2013 on Radiation Protection and Safety in Nuclear Power Utilization (2013).
[7] The Indonesia Civil Code 1320 on Contract Law.

An Analysis of Aircrew Exposure from Cosmic Radiation on Indonesian Domestic Flight

I Made Ardana[1, a)], Aris Sanyoto[1, b)], and Kristyo Rumboko[1, c)]

[1]*Directorate of Regulation Development for Radiation Facilities and Radioactive Materials*
Nuclear Energy Regulatory Agency of Indonesia

[a)] Corresponding author: i.madeardana@bapeten.go.id
[b)]a.sanyoto@bapeten.go.id
[c)]k.rumboko@bapeten.go.id

Abstract. There have been no studies and/or regulations related to cosmic radiation exposure received by aircrew during domestic flights in Indonesia. The paper presents the dose analysis due to the cosmic radiation to aircrew members which is needed to be used as a reference in considering the IAEA recommendation regarding cosmic radiation exposure for Indonesian aircrew. The study was carried out based on a simulation method using a computer program called CARI-7. This computer program can calculate the dose of cosmic radiation which is received by an adult on an air flight trip at a certain time based on the input variables of the origin and destination airports, flight altitude, flight duration, and duration of time to take-off and landing. In this study, Soekarno-Hatta airport was designated as the airport of origin of flights while the destination airports were the ten biggest cities airport in Indonesia. The results showed that the flight from Jakarta to Palembang was the flight that gave the lowest effective dose of 1.1 uSv and the flight from Jakarta to Makassar was the flight that provided the highest effective dose of 3.9 uSv. Also, the measurement results showed that for the same flight time, the effective dose received was greater consequently with increasing flight altitude. Moreover, the effective dose received during the flight was also influenced by the flight time. At the same altitude, the longer a flight the more effective dose will be received. Based on the analysis result, aircrew who working on Indonesian domestic flights with flight times of 1,050 hours/year, were estimated to get cosmic radiation of 1.4 mSv/year. This value exceeded the annually allowable radiation dose limit for the public.

Keywords: Dose, Cosmic Radiation, Domestic, CARI-7, Dose Limit

INTRODUCTION

Indonesia, the largest archipelago in the world, comprises five major islands, namely Sumatra, Java, Kalimantan, Sulawesi, and Papua and about 30 smaller groups with total area of Indonesia is 1.916.862 km^2 (1). As the largest archipelago, Indonesia with a great number of populations is faced with the enormous challenge in transportation sector. Connectivity by road, rail, or ship will takes time. The immediate and logical answer is aviation. Air transportation is a kind of transportation facility that connects all regions in Indonesia in the fastest way (2,3). Along with the increase in population and welfare of the people year by year, the demand for air transportation services in Indonesia is also increasing (3). The increasing demand for air travel services directly increases the frequency of aircrew to be exposed to cosmic radiation. Cosmic radiation (CR) is formed by Galactic Cosmic Rays (GCR) and solar radiation whereas both of them consist of several types of ionizing radiation from external sources to our planet, which interact with the Earth's magnetic field, as well as with the components of the atmosphere. The composition of the GCR is very heterogeneous, including nuclei (about 98% in total, of which 87% consists of hydrogen, 12% of helium and 1% of heavy nuclei), with a small contribution of electrons and positrons (2%) (4). In addition, the contribution of the solar radiation is composed of protons, electrons, helium nuclei and electromagnetic radiation (5).

CR penetration depends on several factors, including the Earth's magnetic field and the attenuation caused by the atmosphere, such that only a part of the incident CR reaches the earth's surface, irradiating all living things continuously, including human beings. The particles of cosmic radiation collide with atoms in the atmosphere, causing ionization and losing their energy gradually. The intensity of cosmic radiation, as well as its composition and co-products, depend on the altitude, and, at higher altitudes, the level of dose received due to cosmic radiation is greater than that at lower altitudes, as can be seen in Fig. 1. The earth's magnetic field also acts as a shield, deflecting the incident particles on the earth. However, there is a strong dependence of deflection on the latitude of incidence. For instance, near the poles, the dose rate, caused by cosmic rays, is two to three times higher than that in equatorial regions (6). This deflection capacity of the particles is determined by a local characteristic of the geomagnetic field called rigidity cut-off (5).

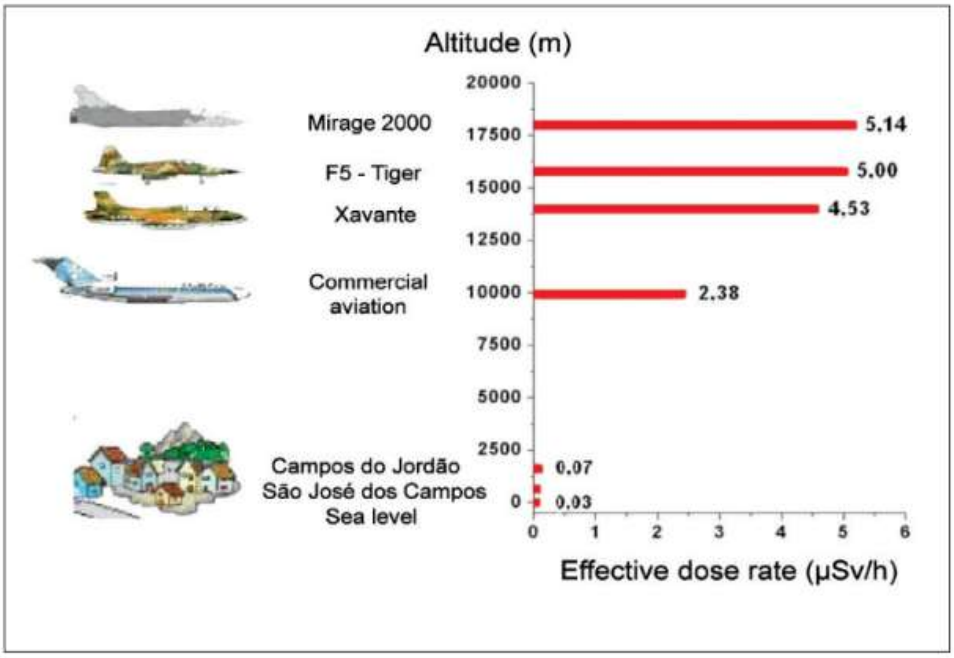


FIGURE 1. Rates of effective dose due to cosmic radiation as a function of altitude [5].

The longer the aircrew works in the air, the greater the potential increase in the dose of cosmic radiation received by the aircrew. This happens because the level of cosmic radiation doses will get higher along with increasing altitude. The dose level of the cosmic radiation is elevated with altitude; it becomes about 100 times higher at the cruising altitude of a commercial jet aircraft than on ground (7). Increasing the dose of cosmic radiation received by aircrew will increase the potential for chromosomal aberration that can trigger aircrew cancer. For an accumulated cosmic radiation dose of 5 mSv per year over a career span of 20 years (a typical prediction for a long haul crew member), the likelihood of developing cancer will be 0.4%. The overall risk of cancer death in the western population is 23%, so the cosmic radiation exposure increases the risk of cancer death from 23% to 23.4%. For a career span of 30 years, the cancer risk increases from 23% to 23.6% (8).

The International Commission on Radiological Protection (ICRP) recommends that exposures to cosmic radiation of aircraft crew be considered as an occupational exposure (9). Following the recommendations of ICRP, the European Union introduced a revised Basic Safety Standards Directive which included cosmic radiation exposure of aircraft crew as occupational exposure. Also, the Government of Japan established a guideline for management of cosmic radiation exposure of aircraft crew. In addition, ICRP recently recommended that frequent flyers also be informed of their dose levels in aviation while their exposure be categorized as public exposure (7). On the other hand, taking into account the IAEA General Safety Requirements Part 3 (GSR Part 3) para 5.30.-5.33 exposure of aircrew and space crew is considered to be existing exposure situation. IAEA GSR Part 3 stated that "The regulatory body or other relevant authority shall determine whether assessment of the exposure of aircrew due to cosmic radiation is warranted. Where such assessment is deemed to be warranted, the regulatory body or other relevant authority shall establish a framework which shall include a reference level of dose and a methodology for the assessment and recording of doses received by aircrew from occupational exposure to cosmic radiation." As a member state, if it is required, Indonesia will certainly try to adopt or adapt the GSR Part 3 recommendation for exposure of aircrew. An initial step to finding out whether the exposure to cosmic radiation received by aircrews on Indonesian domestic flights needs to be monitored or not is the need to analyze the radiation doses from cosmic radiation. Consequently, it is necessary to conduct an analysis of aircrew exposure from cosmic radiation on Indonesian domestic flight. The study was carried out based on a simulation method using a computer program called CARI-7.

METHODS

The complexity of the radiation field to aircrafts flight altitude makes the direct measurement of this field a difficult and expensive work, and few groups dominate this technique in the world (5). Technically, it is possible to measure the dose rate during flight within the airplane. But, measurement of dose for aircraft crew and passengers is difficult, as instruments capable of monitoring the total field spectrum are generally bulky and not very robust (10). Whereas conditions of cosmic radiation are well known, the doses can be sufficiently exact calculated by computer programs. These programs determine the entire effective dose en route, based on physical measurements and flight- determining data (e.g. flight date, departure and destination airport, flight profile and duration). In Germany, three programs are certified by the Federal Office for Aviation for the use of official dose calculation for aircrews (EPCARD, PCAIRE, and FREE). Other Programs that are used in Europe are CARI and SIEVERT (11). As an initial step, this study used CARI-7A as an application in calculating the cosmic radiation dose received by the aircrew of Indonesian domestic flight. In this study, CARI-7A code was used in order to perform dose estimate calculations. The code provide the results in terms of the effective dose quantity, which is a limiting quantity, that is, an appropriate quantity to estimate human health risk due to ionizing radiation and can be directly related to the dose limits set by Nuclear Energy Regulatory Agency of Indonesia (BAPETEN).

CARI-7A calculates the dose of galactic cosmic radiation received by an adult on a nonstop aircraft flight on any date from January 1958 to the present. It can also calculate the effective dose rate from galactic cosmic radiation (GCR) at any specific location in the atmosphere at altitudes up to 100 km for these dates. This computer code has been verified by comparing the calculation result with the measurement result directly using TEPC measurements and modeling with other computer codes such as EXPACS, NAIRAS, and CARI-6. Considering the differences in transport codes and dose calculation techniques, the results show surprisingly good agreement at the most common commercial flight altitudes. Also, this computer code has been validated with the most commonly used computer code for nuclear transport modeling, MCNP. A window display of operated CARI-7A program is shown in **FIGURE 2**. The program takes into account the effects of solar activity, as well as the geomagnetic field on galactic cosmic radiation levels for the date selected by the user. Doses and dose rates are integrated from databases of cosmic ray showers calculated by MCNPX 2.7.0. The shower intensities are derived from the primary cosmic ray (GCR) input spectrum. CARI-7A allows the user to select from multiple preinstalled GCR models and Solar Proton Event models, to use a user defined spectrum (12). As an addition, the program takes into account changes in altitude and geographic location during the course of a flight, based on information provided by the user. For monthly average calculations, databases are used to account for effects of changes in the earth's magnetic field and solar activity on galactic radiation levels in the atmosphere. Flights may also be specified for specific hours of specific days, though this will increase calculation times to accommodate adjustments for geomagnetic storms and forbush effects, if any, on GCR levels at during the flight. The estimated uncertainty from all sources is about 30% for commercial altitudes. The user is required to input the date of the flight, the origin and destination airports (according to the International Civil Aviation Organization/ICAO code), the altitudes and duration of flight at those altitudes. Both the input and calculation results are recorded in a text file.

The study began with determining the flight time, January 1, 2019. There were 10 major airports determined as the origin and destination airports in Indonesia including: Soekarno-Hatta Airport (WIII-Cengkareng), Ngurah Rai (WADD-Denpasar), Kuala Namu (WIMM-Medan), Syamsudin Noor (WAOO-Banjarmasin), Supadio (WIOO-Pontianak), Sultan Mahmud Badaruddin II (WIPP-Palembang), Minangkabau (WIEE-Padang), Sultan Hasanuddin (WAAA-Makassar), Juanda (WARR) -Surabaya), and Adisutjipto (WAHH-Yogyakarta). The International Civil Aviation Organization (ICAO) airport codes are used on this program. Soekarno-Hatta Airport, the biggest and busiest airport in Indonesia (13,14), was designated as the origin airport, while the other airports were varied as destination airports. After determining the flight time, airport of origin, and destination of the airport, an input program was arranged in the form of a TXT file to perform calculations with the CARI-7 program. The number of en route altitudes, minutes climbing to 1st en route altitude, en route altitudes, and minutes descending to destination airport were collected from Flightradar and Flightaware data. As an example, the input file on how to calculate the radiation dose that is received by the aircrew from Soekarno-Hatta Airport to Ngurah Rai Airport is written in appendix 1. As an addition, the CARI-7 program operation steps are shown in **FIGURE 3**. Based on the simulation results, the data of aircrew's effective dose during a single flight route were gained. Then, those data were analyzed to assess the annual effective dose of the aircrew. The analysis was carried out by finding the average hourly effective dose for each flight. Then, the average hourly effective dose is multiplied by the maximum number of flight hours allowed for the aircrew. The average effective dose for aircrew who work by optimizing their flight permits (1,050 hours) then calculated.

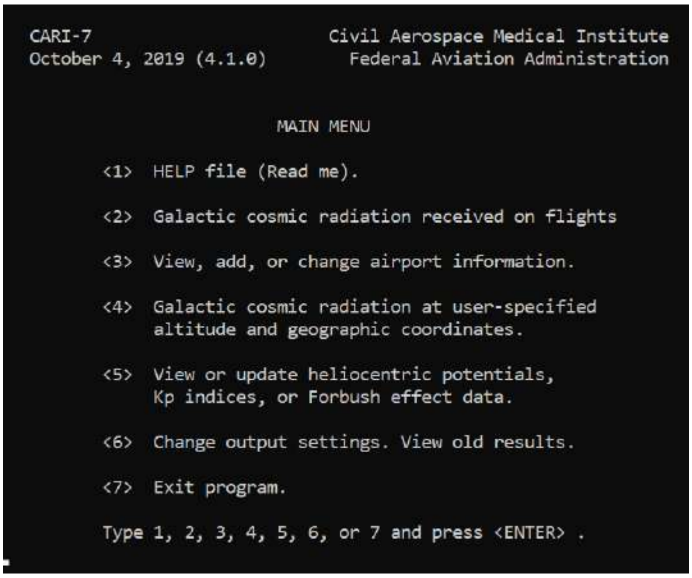


FIGURE 2. A window display of operated CARI-7A program

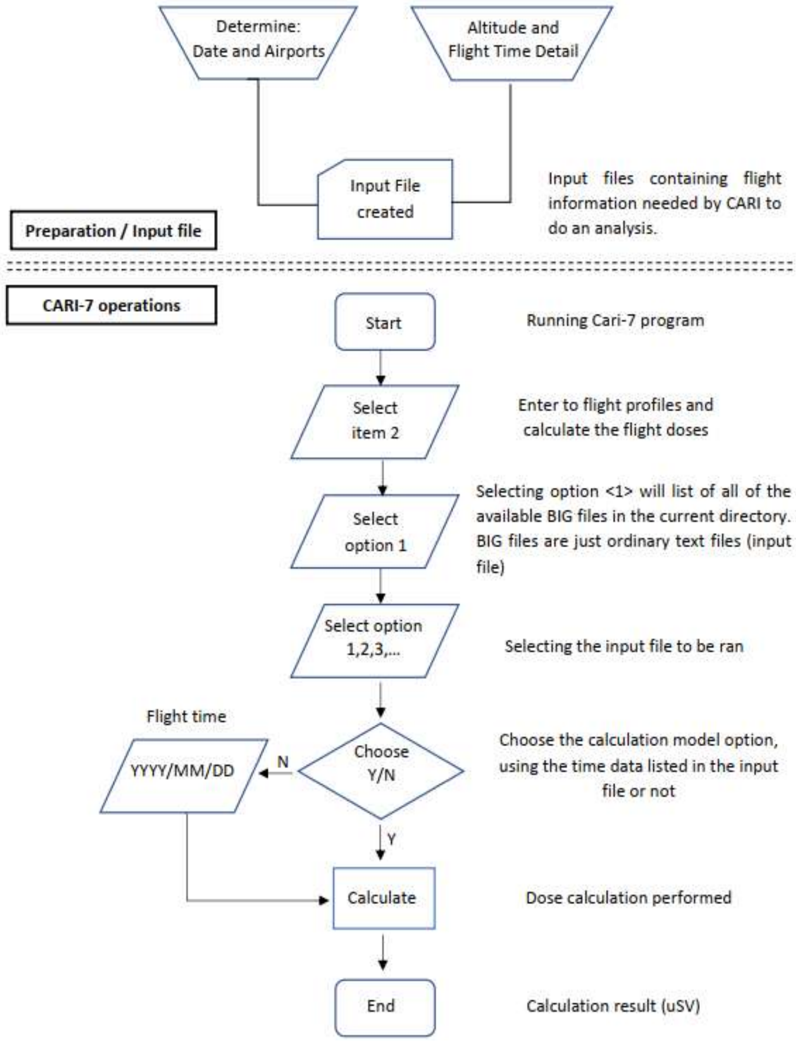


FIGURE 3. the CARI-7 program operation steps to calculate effective dose of aircrew

RESULTS AND DISCUSSION

Since the cosmic radiation increases with increasing altitude, it could be expected that people living/working at high altitudes suffer more from cosmic rays than those at sea level. The intensity of cosmic radiation dose has a much greater magnitude when compared to natural radiation in the ground (15). Based on the study, the effective dose data received by Indonesian domestic flight aircrew during the flight has been obtained as presented in **TABLE 1**. The flight from Jakarta to Palembang provided the lowest effective dose of 1.1 uSv. On the other hand, the flight from Jakarta to Makassar provided the highest effective dose of 3.9 uSv. The calculation results show that for the same flight time, the effective dose received will be greater along with the altitude increment. This happens because the higher the flying area, the energy carried out by the particles of the cosmic rays also be greater. It will cause a greater effective dose. Also, the effective dose received during flight is influenced by flight time. At the same altitude, the longer the flight, the greater the effective dose will be received. As an addition, the geomagnetic latitude also influence the cosmic ray dose rate. As the result of the previous study, the cosmic ray dose rate around the area of 5^0 (north latitude) has the lowest value. This cosmic radiation dose rate will increase as latitude changes to the polar region. The study results which is shown in **TABLE 1** meet all the characteristics of the cosmic radiation dose in the Indonesian airspace.

Based on the analysis results, the average effective dose received by the aircrew is 1.3 ± 0.17 uSv per hour. If an aircrew reaches a maximum limit of allowable flight hours, which is 1,050 hours per year (CASR Part 121), aircrew potentially receives an annual dose effective of 1.4 mSv per year. Those value is far below the value of ICRP recommendation for occupational exposure to ionizing radiation, includes aircrew, which maximum mean body effective dose limits of 20 mSv per year (averaged over 5 years, with a maximum in any 1 year of 50 mSv). But it exceeds the annual radiation dose limit for the public as determined by BAPETEN which is equal to 1 mSv per year. That average annual cosmic radiation dose is the same as the dose received by aircrew in the Netherlands. But, it is lower than the average cosmic radiation dose received by aircrew in other EU countries (11). However, this value is far below from the annual radiation dose limit permitted for worker radiation of 20 mSv per year. Special attention needs to be paid to the cabin crew who usually young women. Pregnant workers not allowed to receive radiation more than or equal to 1 mSv per year to ensure the health of their fetus. Although radiation protection implementation in aircraft's almost inapplicable, several radiation protection and safety scenarios for aircrew must be prepared.

TABLE 1. Effective dose received during flight

No.	CITY		Flight Altitude (ft)	Flight time	Effective Dose (uSv)
	ORIGIN	DESTINATION			
1	JAKARTA (CGK)	BANJARMASIN (BDJ)	35000	01:55	2,5
2	JAKARTA (CGK)	DENPASAR (DPS)	40000	01:55	2,6
3	JAKARTA (CGK)	MEDAN (KNO)	35000	02:25	3,3
4	JAKARTA (CGK)	PADANG (PDG)	35000	01:55	2,6
5	JAKARTA (CGK)	PALEMBANG (PLM)	33000	01:10	1,1
6	JAKARTA (CGK)	PEKANBARU (PKU)	39000	01:50	2,3
7	JAKARTA (CGK)	PONTIANAK (PNK)	35000	01:40	2,1
8	JAKARTA (CGK)	SURABAYA (SUB)	39000	01:40	2,2
9	JAKARTA (CGK)	UJUNG PANDANG (UPG)	40000	02:40	3,9
10	JAKARTA (CGK)	YOGYAKARTA (JOG)	32000	01:20	1,3

Considering the dose of cosmic radiation received by aircrew exceeds the radiation dose limit for the public, BAPETEN should conduct a study to determine whether the supervision of the aircrew exposure is needed or not. Also, cabin crew who are usually dominated by young women should get adequate information related to radiation protection and safety for pregnant women. In accordance with BAPETEN Chairman Regulation No. 4 of 2013 concerning in Radiation Protection and Safety in Nuclear Energy Utilization, licensees are prohibited from placing female radiation workers in working areas that potentially to give radiation doses of more than or equal to 1 mSv/year. Aircrew should report their pregnancy immediately, especially early in the pregnancy period (between the 2nd week until the 18th week) because at that time the fetus is very sensitive to radiation.

For additional information, the European Atomic Energy Community (EURATOM) through the Council Directive 2013/59 / EURATOM specifically in Article 35 paragraph 3 directs member countries to assign flight operators to make arrangements for aircrew who have the potential to receive radiation doses exceeding 1 mSv/year. The intended arrangement can be done through the analysis of doses received by aircrew, taking into account the results of radiation dose analysis in determining the flight schedules, informing the aircrew regarding potential radiation hazards that may be received during work, and paying special attention for female aircrew who are in pregnancy (16). As soon as the pregnancy statement is issued, the worker is prohibited from working in areas that potentially give radiation doses exceeding or equal to 1 mSv/year. Indonesian government may obliged the airline operators to calculate, record, and report the individual effective doses of every aircrew member who may exceed an occupational dose of one 1 mSv/y from cosmic radiation. The calculation of the cosmic ray dose may conducted by a direct measurement using portable detectors or by modelling and simulation methods using computer codes. As an example, in Germany, 45 airlines of various kind (scheduled or charter flights, air cargo, business jets, military etc.) calculate route doses of their personnel with computer programs and transmit the accumulated monthly doses though the Federal Office of Aviation to the Radiation Protection Register of the Federal Office for Radiation Protection (11).

CONCLUSION AND REMARKS

Based on the results of the study, the estimated cosmic radiation dose received by aircrew on flights from Cengkareng to Palembang, which took 1 hour 10 minutes to fly, was the flight with the lowest cosmic radiation dose of 1.1 uSv. The highest dose of cosmic radiation in this study was obtained on flights from Cengkareng to Makassar with an effective dose of 3.9 uSv which took 2 hours 40 minutes to fly. Based on the analysis results, the average effective dose received by the aircrew is 1.3 ± 0.17 uSv per hour. Indonesian aircrew who work by optimizing their flight permits (1,050 hours), are estimated to obtain cosmic radiation of 1.4 mSv per year. The author recommends further analysis of cosmic radiation exposure received by aircrew using other computer programs such as SIEVERT, EPCARD, and PCAIRE by adding variable effects from storms/solar winds so that it can produce more comprehensive data. Measurement of cosmic radiation dose directly with radiation monitoring devices will also be able to improve the quality of research on this topic. The research conducted can be used as a reference in determining the urgency of BAPETEN's supervision arrangements for cosmic radiation dose received by the Indonesian aircrew. The estimated effective dose received by aircrew that exceeds the value of 1 mSv can be one consideration for including aircrew as a part of radiation workers even though the aircrew do not work in radiation facilities. In addition, the effective dose will naturally increase if the analysis is carried out on aircrew who working on international flights. Finally, Nuclear Energy Regulatory Agency (BAPETEN) and Directorate General of Civil Aviation - Ministry of Transportation need to issued recommendations or standards in this respect yet, following the international recommendations.

ACKNOWLEDGEMENT

We thank to the Director of Regulation Development for Radiation Facilities- BAPETEN, Ms. Taruniyati Handayani for the opportunity to the authors to conduct this study and publish the results.

REFERENCES

[1] BPS. Statistik Indonesia Dalam Infografis 2019. 2019;

[2] U.S Embassy Jakarta. The Aviation Sector – Opportunities in Indonesia. 2012;(March 2012).

[3] BPS. Statistik Transportasi Udara 2018. 2018.

[4] Barlett DT, Beck P, Bilski P, Bottollier-Depois JF, Lindborg L. Cosmic Radiation Exposure of Aiecraft Crew. Vol. 1, Radiation Protection 140. 2004. 281 p.

[5] Federico CA, de Castilho Pereira HH, Pereira MA, Gonçalez OL, Caldas LVE. Estimates of cosmic radiation dose received by aircrew of DCTA's flight test special group. J Aerosp Technol Manag. 2010;2(2):137–44.

[6] Lewis BJ, Bennett LGI, Green AR, Butler A, Desormeaux M, Kitching F, et al. Aircrew dosimetry using the Predictive Code for Aircrew Radiation Exposure (PCAIRE). Radiat Prot Dosimetry. 2005;116(1–4):320–6.

[7] Yasuda H, Yajima K. Verification of cosmic neutron doses in long-haul flights from Japan. Radiat Meas [Internet]. 2018;119(August):6–11. Available from: https://doi.org/10.1016/j.radmeas.2018.08.016

[8] Bagshaw M. Cosmic Radiation in Commercial Aviation. King's Coll London. 2007;

[9] Benninson D, Jammet H, Sinclair WK, Meinhold CB, Liniecki J, Dunster HJ, et al. ICRP Publication 60 Recommendation of ICRP 1990. 1st ed. Oxford-New York-Frankfurt-Seoul-Sydney-Tokyo: Pergamon

Press; 1991.
[10] Lochard J, Bartlett DT, Rühm W, Yasuda H, Bottollier-Depois JF. ICRP Publication 132: Radiological Protection from Cosmic Radiation in Aviation. Ann ICRP. 2016;45(1):5–48.
[11] Frasch G, Stegemann R, Kammerer L. Aircrew Monitoring of Occupational Exposure to Ionising Radiation. 2005;(May 1996):1–6.
[12] Copeland K. Cari-7A: Development and validation. Radiat Prot Dosimetry. 2017;175(4):419–31.
[13] Pradana AN, Ahyudanari E, Istiar I. Evaluasi Ketersediaan Gate Di Terminal 3 Ultimate Bandar Udara Internasional Soekarno-Hatta. J Tek ITS. 2017;6(1):3–7.
[14] 1Wicaksana AA, Effendi FM, Warsito T. Slot Time Capacity in Peak Hour At Soekarno-Hatta International Airport. 2018;147(Grost 2017):650–60.
[15] Tobiska WK, Meier MM, Matthiae D, Copeland K. Characterizing the variation in atmospheric radiation at aviation altitudes [Internet]. Extreme Events in Geospace: Origins, Predictability, and Consequences. Elsevier Inc.; 2018. 453–471 p. Available from: http://dx.doi.org/10.1016/B978-0-12-812700-1.00018-2
[16] EURATOM. Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom a. Off J Eur Commun L13. 2014;(December 2003):1–73.

APPENDIX

Format of an input file which is consists of a flight profile in a big file to calculate effective dose received by aircrew which is flies from CGK to DPS on January 1, 2019.

Input	Info
CGK-DPS	Flight information (from Cengkareng to Denpasar)
01/2019	Flight date (MM/YYYY)
WIII	ICAO code of origin airport (Soekarno-Hatta)
WADD	ICAO code of destination airport (Ngura Rai)
2	Number of en route altitudes
25	Minutes climbing to 1st en route altitude
36500 25	1st en route altitude: feet minutes
35000 30	2nd en route altitude: feet minutes
20	Minutes descending to destination airport

Natural Radioactive Discrimination to Determine Air Contamination Levels of Workplaces

Nofriady Aziz[1,a)] Nudia Barenzani[1,a)], Anang Setiawan[1,a)], Arca Datam S[1,a)]

[1]*Pusat Teknologi Bahan Bakar Nuklir – BATAN*

Email: [a)] nofriady-aziz@batan.go.id

Abstract. The existence of natural radioactivity that is measured in the footage must be discriminated (separated) by utilizing the differences in the nature of the half-life of both radioactivity, namely by delaying the counting of a few moments to give the chance of decaying radioactive nature so that it can be ignored. The purpose of this study was to determine air contamination caused by Uranium decay after discrimination against radioactive nature. This research was carried out by sampling air, contaminants and air with contaminants. The highest alpha air contaminant activity is found in room's 5 which is equal to 0.022 Bq/m3. All results of the calculation of air contamination are still below the threshold set by MPC (Maximum Permissible Concentration) which is under 2 Bq/m3. Keywords: Nature Radioactive, Air Contaminant, MPC

INTRODUCTION

Monitoring of the Element Fuel Experimental Installation (EFEI) work facility (laboratory) area is carried out so that the activities at the work facility (laboratory) at EFEI run securely and safely for workers, the public and the environment against the dangers of radiation and contamination. Monitoring of work areas carried out includes: monitoring of gamma radiation exposure, derived air contamination and radioactivity contamination level (gross alpha) on the floor / table surface of the work area.[1] Measurement of the level of airborne radioactivity contamination is carried out directly and indirectly. Direct measurements were carried out by using an alpha survey meter in areas potentially radioactive contaminated. Measurements were also made using the gamma survey meter instrument at locations indicated to have exposure that exceeds background level significantly. Measurement of the level of contamination indirectly includes several stages of activity namely sampling, counting of sample, calculation and evaluation of the calculation results.[2] The purpose of this study is to determine air contamination caused by Uranium decay after discrimination against natural radioactivity.

Monitoring the level of airborne radioactivity contamination in the working room of a nuclear installation is influenced by the presence of natural radioactive substances that are also present in the air so that they are sampled and held in the filter paper when taking air samples. The potential for air contamination caused by the activities carried out in the laboratory comes from uranium. However, the uranium decays to form Radon elements in the form of Rn-220 gas. [3]

Rn-220 is also called Thoron. Gas - the gas diffuses continuously through the building wall surface-to-air installation work room. [4] In the air Radon and Thoron decays to form their whole daughters into a stable nuclide including Ra-B (Pb-214) with a half-life of 26.8 minutes, Ra-C (Bi-214) with a half-life of 19.7 minutes; Th-B (Pb-212) with a half-life of 10.6 hours; Th-C (Bi-212) with a half-life of 60.6 minutes during air sampling Radon and Thoron which are not sampled but their decay daughters.[5], [6]

To determine the level of contamination in the workspace air caused by radioactive contaminants of uranium nuclear material, then in analyzing air samples they discriminated against Radon and Thoron daughters. The discrimination is carried out by delaying the enumeration of some footage so that the Radon daughter decays and can be ignored. The level of contamination is determined using the following equation.[5]

$$A(to) = A_{Tn}(to) + A_{Rn}(to) + A_K(to) \tag{1}$$

In the analysis of air sample, it is assumed that 1/500 of the initial activity of Radon daughters can be neglected ($A_{Rn}(to)$=0), hence counting of air samples for 4 hours (at time t1) is:[5]

$$A(t1) = A_{Tn}(t1) + A_K(t1) \tag{2}$$

During the time interval $A_{Tn}(t1)$ will decay by the decay factor $e^{-\lambda_{Tn}\Delta t}$and $A_K(t1)$ will decay by the decay factor $e^{-\lambda_K \Delta t}$ so that the aerial decay factor at t2 becomes Equation 3[5]

$$A(t2) = A_{Tn}(t1)e^{-\lambda_{Tn}\Delta t} + A_K(t1)e^{-\lambda_K \Delta t} \tag{3}$$

Equation 3 is substituted into Equation 2; it would be Equation 4 [5]

$$A_K = \frac{A(t2) - A(t1)e^{-\lambda_{Tn}\Delta t}}{e^{-\lambda_K \Delta t} - e^{-\lambda_{Tn}\Delta t}} \qquad (4)$$

The radioactive material handled at EFEI is Uranium which has a half-life for U-238 which is 4.51 x 109 years and U-235 which is 7.1 x 108 years. So if the life time of air contaminants at t2 is 24 hours, then the decay of contaminant activity during the analysis of air samples can be ignored ($\lambda_K \Delta t = 0$) so that the contaminant decay factor is worth 1 ($e^{-\lambda_K \Delta t} = 1$). Equation 4 can be simplified to Equation 5 [5]

$$A_K = \frac{A(t2) - A(t1)e^{-\lambda_{Tn}\Delta t}}{1 - e^{-\lambda_{Tn}\Delta t}} \qquad (5)$$

Thoron daughter (Th-B) has the longest half-life that is 10.6 hours and a decay constant of 0.0654/hour. If Δt = 20 hours (t1 = 4 hours and t2 = 24 hours), then by entering the number in Equation 5, Equation 6 will be obtained [5]

$$A_K = \frac{A(t2) - 0{,}2705A(t1)}{0{,}7295} \qquad (6)$$

Note: Ak = contaminant activity
$A(t_1)$ = sample activity when counting (sample life) at 4 hours
$A(t_2)$ = sample activity when counting (sample life) at 24 hours

Evaluation is an activity that compares the results of the procedure with the criteria and standards that have been set to see its suitability, then it is available information about the extent to which a certain activity has been achieved, so that it can be known if there is a difference between the standards set and the results that can be achieved. To achieve the evaluation objectives, a comparison will be made between the results of monitoring activities in the laboratory with the safety provisions set by the Regulatory Body, in this case BAPETEN (**TABLE 1**).

Table 1. Maximum Permissible Concentration / MPC [2,7]

Zone	Air Contamination Level
	Radioactivity α
I	Background
II	<2 Bq/m^3
III	<20 Bq/m^3

METHODOLOGY

The equipment used for this activity is an air sampling tool that functions to collect radioactive substances that are dispersed in the workspace air through filter paper and a radiation counter to count the radiation collected on filter paper (air samples), such as shown in **FIGURE 1.**

(a)

(b)

FIGURE 1. (a) Air Sample (b) Radiation Counter

In this evaluation activity, the monitoring results data evaluated are limited to the working area of the PTBBN laboratory.

Contaminant Sampling Procedure

Contaminants that contain uranium which have a long half-life are dropped on filter paper and are counted the contaminants with an Alpha Beta sample counter.[9]

Sampling Natural Radioactivity Procedure

Measurements of natural radioactivity were carried out in the morning, then GF-8 filter paper was installed in the air sampler for 30 minutes with an air flow of 30.4 L/m and counts of contaminants with Alpha Beta sample counter.[8]

Sampling natural radioactivity and contaminants procedure

The contaminants that have been chopped are then placed in an air sampler for 30 minutes with an air flow of 30.4 L / m3 and counted for the contaminants with an Alpha Beta sample counter. [8] [9]

Air sampling of PTBBN's laboratory work area procedure

The GF-8 filter paper is installed in the air sample filter holder then the air sampler is operated for 30 minutes with an air discharge of 30.4 L / m3. Air sampling is done in some places or workspaces with the potential for air contamination. Filter paper is counted with the Alpha Beta sample counter immediately after sampling as an air radioactivity. However, for the purpose of determining the level of air contamination in the workspace, the counting is carried out after a delay of 4 hours, then the counting is carried out after the 24 hours sampling time.[8] [9]

To determine the activity on natural radioactivity, the combination of natural radioactivity with contaminants and air sampling activities can be calculated with Equation 7 while to determine the activity of contaminants can be calculated with Equation 8: [8]

$$A_u = C\ x\ FK\ x\ \frac{1}{d} x \frac{1}{t} \tag{7}$$

$$A_c = C\ x\ FK \tag{8}$$

with:

Au	=	activity of radioactive substances in the air (Bq/m^3);
Ac	=	Activity of air contaminants (Bq/m^3);
C	=	Count per second (cps);
FK	=	Conversion factor: 1,2 Bq/cps
D	=	air suction discharge (m^3/minute);
T	=	air suction discharge (minute).

RESULT AND DISCUSSION

Measurement Air Radioactivity

This measurement of air radioactivity is carried out before the VAC is operated at 30 minutes intervals until natural radioactivity can be ignored (1/500). Measurement data is shown in **TABLE 2**. In **TABLE 2** it can be seen that the greater the time of counting, the amount will be smaller and the activity also (Bq/m3) the smaller this is caused by the decay of Thoron in the air will decay up to 4 hours (240 minutes) marked at 255 minutes, the count began to stabilize.

Table 2. Measurement Air Radioactivity

No	Time (minute)	Count (cps)	Activity (Bq/m^3)	No	Time (minute)	Count (cps)	Activity (Bq/m^3)
1	0	15.842	38.253	10	135	2.758	1.812
2	15	13.408	32.377	11	150	1.775	1.166
3	30	10.942	26.421	12	165	1.442	0.947
4	45	9.083	21.934	13	180	1.025	0.673
5	60	7.292	17.607	14	195	0.867	0.569
6	75	6.517	15.736	15	210	0.692	0.454
7	90	5.433	13.120	16	225	0.433	0.285
8	105	4.808	11.611	17	240	0.342	0.224
9	120	3.492	2.294	18	255	0.367	0.241

Measurement of Air with Contaminants

Measurement of air with contaminants is done by measuring the radioactivity of air on filter paper dropped by contaminants. Before measuring air with contaminants, the measurement of contaminant radioactivity is carried out to compare the results of air measurements with contaminants. The measurement results are shown in **TABLE 3**. In **TABLE 3** it can be seen that the greater the time of counting, the count will be smaller and the activity will also be (Bq/m3) the smaller. That happens because to the decay of Thoron daughters will decay up to 4 hours (240 minutes) and only contaminants with a long half-life are left. The number of contaminant counts can be seen in **TABLE 4**.

TABLE 3. Measurement of air with contaminants

No	Time (minute)	Count (cps)	Activity (Bq/m^3)	No	Time (minute)	Count (cps)	Activity (Bq/m^3)
1	0	31,942	20,981	10	135	3,925	9,478
2	15	25,058	16,460	11	150	3,458	8,351
3	30	20,200	13,269	12	165	3,367	8,130
4	45	15,558	10,220	13	180	2,850	6,882
5	60	11,408	7,494	14	195	2,875	6,942
6	75	8,433	5,540	15	210	3,400	8,210
7	90	5,650	3,711	16	225	3,208	7,747
8	105	5,317	3,492	17	240	**2,975**	7,184
9	120	4,083	9,860	18	255	**3,100**	7,486

TABLE 4. Measurement contaminants radioactivity

No	Time (minute)	Count (cps)	No	Time (minute)	Count (cps)
1	0	2,458	10	135	2,933
2	15	2,733	11	150	2,908
3	30	2,883	12	165	2,775
4	45	3,008	13	180	3,000
5	60	3,050	14	195	2,942
6	75	3,133	15	210	2,958
7	90	3,033	16	225	3,000
8	105	2,900	17	240	2,975
9	120	2,950	18	255	2,983

Based on measurements of air with contamination and without contaminants it can be illustrated that the decay of the Thoron daughters will occur as shown in **FIGURE 2**.

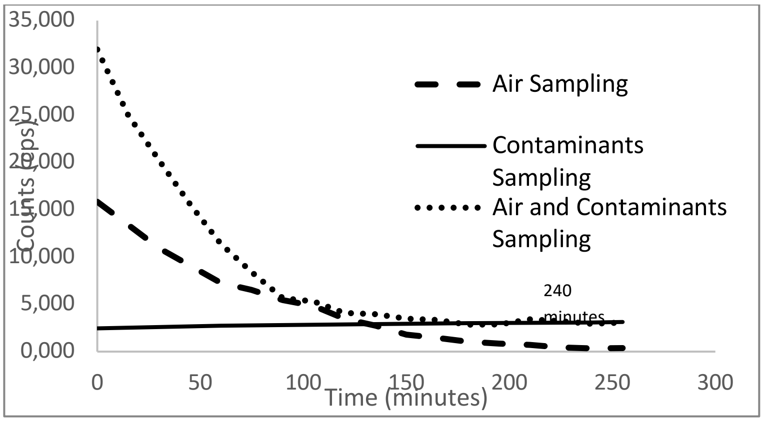


FIGURE 2. Measurement of Air Radioactivity, Contaminants, and Air with Contaminants

In **FIGURE 2**, it can be showed that natural air activity will decay after an interval of 240 minutes (4 hours) due to the presence of a lot of Radon gas and Thoron gas in the form of its decay Radium-B (Ra-B). Thoron with a half-life (T½ Th-B 10.6 hours) decays into a stable nuclide including Ra-B (Pb-214) with a half-life of 26.8 minutes, Ra-C (Bi-214) with a half-life of 19.7 minute; Th-B (Pb-212) with a half-life of 10.6 hours; Th-C (Bi-212) with a half-life of 60.6 minutes so that it was sampled during air sampling and this phenomenon caused the activity of air contamination to be high as shown in **FIGURE 2**. Radionuclide activity in samples will decrease and become stable after decay from Radon and Thoron gas. Data from the measurements of contaminant activity shown in **TABLE 3** look relatively constant or stable, this is caused by the half-life of the isotope U as a long contaminant (T½ U-238, 4.51 x 109 years and U-235, 7.1 x 108 years).

Sampling Work Room Facility

In this activity carried out to determine air contamination in the working facility by measuring air radioactivity at intervals of 4 hours and 24 hours and using Equation 6. The measurements were carried out in the EFEI facility laboratory with the highest alpha radiation. After measuring, the results obtained in **TABLE 5**.

TABLE 5. Measurement radioactivity in work room

Room	Alpha			
	A_o (Bq/m^3)	At_1 (Bq/m^3)	At_2 (Bq/m^3)	Ak ((Bq/m^3)
Room 1	1,174	0,060	0,040	0,018
Room 2	1,630	0,074	0,034	0,006
Room 3	1,261	0,080	0,040	0,010
Room 4	0,872	0,101	0,047	0,010
Room 5	1,040	0,121	0,067	0,022

In **TABLE 5** it can be seen that the initial activity (Ao) has a large activity caused by Radon and Thoron daughters, the activity after 4 hours (A (t1)) activity begins to decrease and by delaying for 24 hours (A (t2)) then the activity will be reduced more and by using Equation 6 can be calculated contaminant activity (At). Measurements were taken in room 1, room 2, room 3, room 4 and room 5. Room 1 until Room 5 exist in EFEI which is Fabrication Fuel Laboratory. The highest alpha air contaminant activity was found in Room 5 which was 0.022 Bq/m3 but was still below the threshold set by the MPC at 20 Bq/m3 after a comparison between the results of monitoring activities in the laboratory with the safety provisions by BAPETEN.

CONCLUSION

Natural air activity will decay after an interval of 240 minutes (4 hours). The radioactivity after 4 hours will decrease and by delaying for 24 hours the activity will decrease more. The highest activity of airborne radioactivity contamination is in room 5 which is 0.022 Bq/m3 but is still below the threshold set by the MPC of 20 Bq/m3.

ACKNOWLEDGMENTS

Authors say thanks to PTBBN and those who have assisted in this research activity.

REFERENCES

[1] Nudia Barenzani, Nofriady Aziz, Arca Datam Sugiarto, dan Nur Yulianto Darojad. 2018. Penentuan Mulai Waktu Kerja Di Laboratorium IEBE Berdasar Pengukuran Keradioaktifan Udara. Yogyakarta. Prosiding SDMTN 2018
[2] PTBBN, Laporan Analisis Keselamatan Instalasi IEBE, No. Dok: KK20J09002, Rev.7, 2012
[3] ICRP. 1987. Lung Cancer Risks from Indoor Exposures to Radon Daugh ters ICRP Publication 50.
[4] The World Health Organization. 2009. WHO handbook on indoor radon: a public health perspective. Geneva: WHO.
[5] Lasuk, S.R., et.al, "Radiation Safety Techinisian Training Course", ANL-7291 Rev.1 : Health and Safety, ANL, Illionis, 1972.
[6] IAEA, 'Radiological Surveillance of Airborne Contaminants in the Working Environment', Safety Series No.49, IAEA, Vienna, 1978.
[7] Perka BAPETEN No. 4, Tahun 2013 tentang Proteksi dan Keselamatan Radiasi dalam Pemanfaatan Tenaga Nuklir
[8] Sjafruddin, dkk. 2018. "Penentuan Tingkat Keradioaktifan Kontaminasi Udara Ruangan Kerja Di IRM". Jakarta. PTBBN.
[9] PTBBN, SOP Pemantauan Paparan Radiasi dan Keradioaktifan Daerah Kerja di IRM, No. Dok.: SOP 024.002/KN 04 02/BBN 5.1.

The Effect of Water Flow Rate Against Water Conductivity at Water Treatment System Merah-Putih Gamma Irradiator

Arif Rachmanto [1, a)] and Bimo Saputro [1, b)]

[1]The Center for Isotope and Radiation Application, Lebak Bulus Street No.49 South Jakarta

[a)] Corresponding author: arif_rachmanto@batan.go.id
[b)] another author: bimo_saputra@batan.go.id

Abstract. Resin in the water treatment system of the Merah-Putih Gamma Irradiator (IGMP) is used as ion exchange. The pool containing demineralized water with low conductivity, less than 10 µS, to prevent metal components installed at the bottom part of the irradiator from corrosion. However, evaporation mainly due to heat from radiation source causes water level is reduced. Raw water from other resources must be added to maintain a demineralized water level. Theoretically, the conductivity depends on the contact area and contact time between the raw water with resin particles. Calculation from such a situation is difficult to be realized. Therefore, the conductivity and pH of the raw water is measured after being the raw water is reacted with the resin particles in the water treatment system. The exchange process is influenced by several factors, including the water flow rate. The purpose of this research is to find out the effect of water flow rate against conductivity so that the right water flow rate caused the ion exchange process runs optimally can be determined. The water treatment system is operated in filling mode. In the filling process, the water filling flow rate is varied in six, 100 liters/hour, 200 liters/hour, 300 liters/hour, 400 liters/hour, 500 liters/hour, and 600 liters/hour, respectively. Each water filling flow rate was observed for conductivity and water pH after passing through cation resin and anion resin. Data is collected every 3 minutes. Based on the datasheet, the optimal water flow rate through the cation resin is 120 m^3/hour and the optimal water flow through the anion resin is 60 m^3/hour. This research shows if the amount of water flow rate pass resin is closer to the value recommended by the datasheet, it can cause the ion exchange process runs optimally and the expected low conductivity is achieved when the pump was operated at the speed 600 liters/hour which produces 64.3 m^3/hour at the point after passing through the resin tank. However, the pump is recommended to be operated at a maximum speed of 300 liters/hour, otherwise the PVC pipe break or burst. To overcome this situation, design engineering is needed, so that the pump can be operated at the recommended speed.

Keywords: Conductivity, resin, water treatment system, water flow rate, pH

INTRODUCTION

In gamma irradiators, especially it is of category 4, the radiation sources are stored in a pool containing demineralized water if they are not used for irradiation processing. To maintain the water conductivity values below 10 micro-siemens, water is circulated and generated through a water treatment system (WTS) to produce demineralized water. Demineralized water is water that has been processed in such a way that its cation-anion mineral contents are removed. The mineral content as a form of cation and anion in water are including Na^+, Ca^{2+}, Mg^{2+}, K^+, Fe^{3+}, Cl^-, SO_4^{2-}, and CO_3^{2-} [1]. The WTS is also used to supply additional pool water that has evaporated mainly due to heat released by radioactive sources. Demineralized water has a function as a radiation absorber, where the exposure rate on the surface of the pool should not exceed 10 µSv/h. Besides, the gamma irradiator is also equipped with an ultrafiltration system. The use of ultrafiltration in water purification system has significantly increased and it has potential to replace conventional systems. This paper shows that the quality of water in outlet purification and make-up water dual system depends on the nuclear grade conditions. Outlet water quality of cationic and anionic demineralizer placed at the end of the dual system for purification. Moreover, this water is slightly acidic. Also, since the pool water is in contact with the atmosphere, it is saturated with oxygen. Carbon dioxide absorbed from the atmosphere reacts with water form carbonic acid, which tends to make the pH mildly acidic (pH 5.5–6). Sources of potential chemical contaminants at the gamma irradiator storage pool includes airborne materials (dust, etc.), make-up water, and leaching from materials in the pool. These conditions provide good water transparency and corrosion resistance of gamma irradiator cladding and other structures in the storage pool [2]. The need for clean water in the industry is done by analyzing the kinetics of the demineralization process. It is important to know the particle size and the effectiveness of the process to produce clean water [3]. Small particles are preferred to demineralize the processed water. Moreover, the use of nanofiltration membranes, monovalence concentration can reduce permeability [4].

Merah-Putih Gamma Irradiator is one of the irradiation facilities using gamma rays. Merah-Putih Gamma Irradiator is a type 4 irradiator. The radiation source is stored in pool water containing demineralized water when it is not operating. In water treatment systems, ion exchange resin has a function as filters for mineral ions. Filtering processes is performed by exchanging ions contained in raw water with ions in the resin. Resins are divided into two types, namely cation and anion resins. Cation resins function as positive ion exchanges and anion resins as negative ion exchanges. When used, resins have a limit on ion exchangeability so they can make the resin-saturated and no longer optimal in conducting ion exchange. Especially for irradiators with wet storage rooms, they are required to have a mineral free water treatment system, where the water produced must have a conductivity value below 10 micro siemens. Saturated conditions resin when the water conductivity value fluctuates in the circulation process, the conductivity value is more than 10 micro siemens during the process of filling pool water and circulation and the pH value is close to neutral during the process of filling pool water. If resin at saturated conditions, the resin must be regenerated. If after the regeneration process the resin condition is still saturated then the resin cannot be reused. So that the resin must be replaced with a new one maintain pool water quality by standards [5]. Today most ion exchange resins bases are styrene and divinylbenzene which is then sulfonated [6]. Resin has other benefits, one of which is that cation resin is used as a catalyst. For example, Dowex 50 as a catalyst in the fat hydrolysis reaction [7].

Demineralized water has the function as a radiation shield so that the radiation exposure caused by the radiation source does not exceed the background limit. The radiation source used in Merah-Putih Irradiator is Co-60 because it has high energy, has a long half-life (5.27 years), and is insoluble in water [2]. The interaction between the radiation source and demineralized water causes the temperature to rise and evaporation occurs. The evaporation causes the volume of demineralized water in the pool to be reduced. This demineralized water management is carried out by a water treatment system. It is to say that the water treatment system is a facility that cannot be separated from gamma irradiators. The water treatment system has the function of maintaining the conductivity value of pool water based on established standards.

The Water Treatment System facility has three types of operating modes namely circulation, pool water filling, and resin regeneration. The process of circulation is the process of flowing irradiator pool water through an ion exchange resin then flowed back into the pool so that the conductivity is maintained. The process of pool water filling is the process of flowing raw water through an ion exchange resin then the water has flowed into the pool until the water level reaches normal limits. The resin regeneration process is the process of reactivating the ability of the ion exchange resin by flowing with a chemical so that the resin's performance is optimal again [5]. The water treatment system uses an ion exchange resin that has a function as a filter for mineral ions.

The filtering of mineral ions is done by exchanging ions between ions in raw water with ions in the resin. The exchange process is influenced by several factors, including the water flow rate. The purpose of this research is to find out the effect of water flow rate against conductivity so that the right water flow rate caused the ion exchange process runs optimally can be determined. Similar research has been done by previous researchers. The difference in previous research between the research’s author is the object of the research and method. Widarti has researched "The influence of feed flow rate toward the capacity of commercial cation exchanger resin and adsorption of the metal ion with difference valence"[8]. Increased water flow rate causes the chance of metal ions to bond with negative functional reduced. Certainly, the water flow rate is one factor that caused the exchange process to run optimally.

METHODS

Materials and Tools

Materials and tools used in this study were T42 cation resin, A23 anion resin, water treatment system, conductivity meter, and pH meter. Conductivity meter is used to measure the conductivity of water. Ph meter is used to measure the pH of water.

Methods

The steps used to determine the effect of water flow rate on the ion exchange resin of the water treatment system are as follows: The water treatment system is operated in filling mode. In the filling process, the water filling flow rate is varied in six, 100 liters/hour, 200 liters/hour, 300 liters/hour, 400 liters/hour, 500 liters/hour, and 600 liters/hour, respectively. Each water filling flow rate was observed for conductivity and water pH after passing through cation resin and anion resin. Data is collected when the water has been flowed for 3 minutes for an individual predetermined flow rate setting. Based on the datasheet, the optimal water flow rate through the cation resin is 120 m^3/hour and the optimal water flow through the anion resin is 60 m^3/hour.

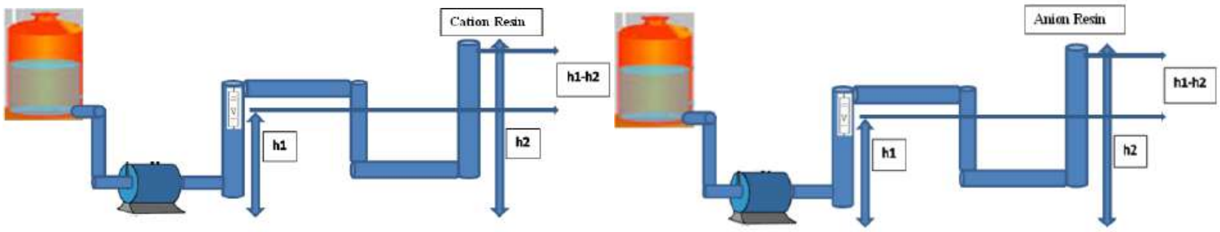
A simple schematic position of the water flow rate after passing the pump (h1) and anion resin and cation (h2) is shown in **FIGURE 1**. A complete schematic is shown in **FIGURE 2**. Because the water flow rate data in **FIGURE 1** is the water flow rate after passing through the pump, the amount of water flowing through the resin must be found. To find the water flow rate that passes through the anion resin (V_2) Bernoulli's law can be used, as follows.

$$P_1 + \frac{1}{2}\rho {V_1}^2 + \rho g h_1 = P_2 + \frac{1}{2}\rho {V_2}^2 + \rho g h_2 \tag{1}$$

$$\text{where } P_2 = \frac{1}{4}P_1 \tag{2}$$

$$V_2 = \sqrt{\frac{2*\left(\frac{3}{4}P_1 + \rho g(h_1 - h_2) + \frac{1}{2}\rho {V_1}^2\right)}{\rho}} \tag{3}$$

Where ρ is the density of water (1000 gr/cm^3), V_1 is the velocity of water after passing the pump (m/s) and P is the water pressure (Pa).



(a) (b)

FIGURE 1. (a) Schematic position of Pump (h1) and cation resin tank (h2) and (b) Schematic position of Pump (h1) and anion resin tank (h2).

Based on the calculation in equation no 3, the water flow rate passed the pump, and the water flow rate after passed resin cation and anion are shown in **TABLE 1**.

TABLE 1. The water flow rate passed the pump and cation and anion resin

No	Flow rate water passed the pump (m^3/hour)	Flow rate water after passed cation and anion resin (liters/hour)
1	100	20
2	200	32
3	300	44
4	400	53
5	500	58
6	600	64

RESULT AND DISCUSSION

There are two things in the discussion of this paper, namely the effect of water flow rate on water conductivity after passing through cation and anion resins and the effect of water flow rate on water pH after passing through cation and anion resins. The first discussion is the effect of water flow rate on the conductivity of water after passing through the resin.

In this discussion, there are two types of water flow rate, they are water flow rate from storage tanks after passing through the pump and water flow rate when passing through the resin. The water flow rate data in

FIGURE 3 is the water flow rate after passing the pump. The graph shows that the greater the water flow rate the smaller the conductivity of water after passing through anion resin. But on the water flow rate data after passing cation resin, the conductivity values are fluctuative. To analyze it, we need an optimal water flow rate data sheet that passes through cation and anion resins.

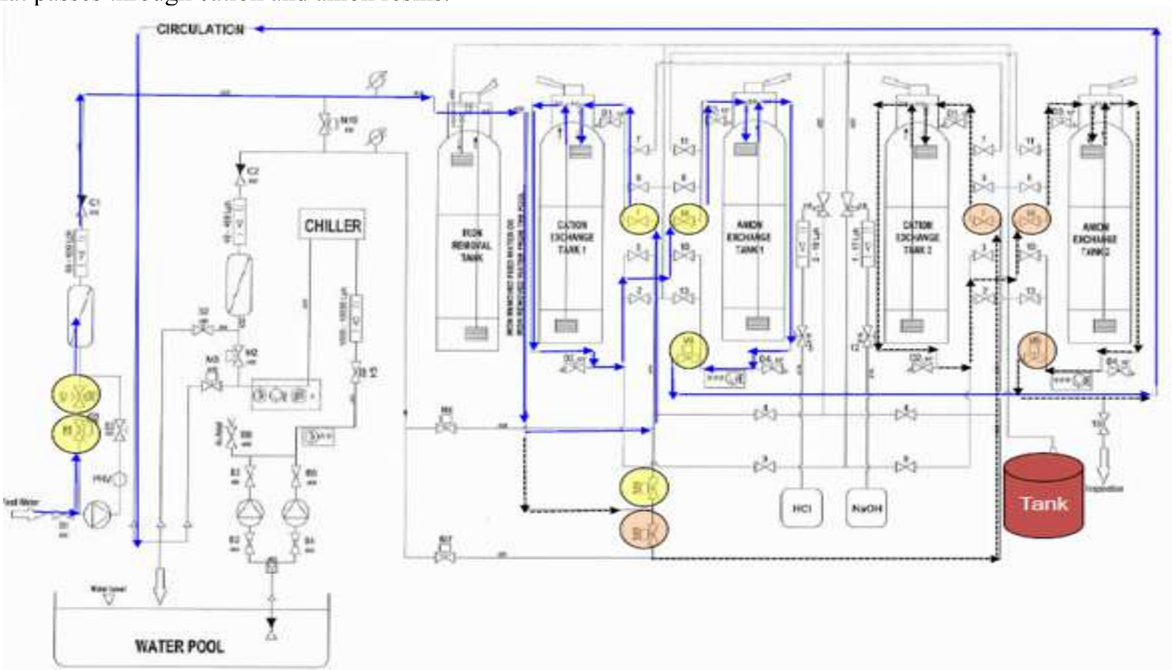
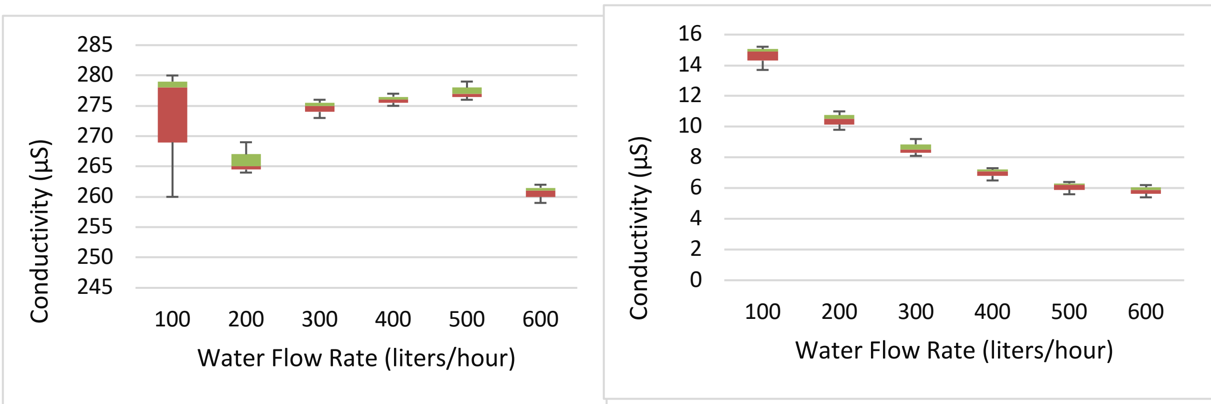


FIGURE 2. Diagram of water flow at the pool filling process



(a) (b)

FIGURE 3. (a) Graphic relationship between water flow rate after through water pump and conductivity after through cation resin and (b) Graphic relationship between water flow rate after through water pump and conductivity after through anion resin

Based on the datasheet, the optimal flow rate for occurring resin exchange at T42 tulsion cation resin is 120 m3/h. The graph in Figure 3 shows that the conductivity of the water after passing through the cation resin begins to decrease at 64 m^3/hr. The water flow rate of 64 m^3/hour occurs when the water flow rate after passing through the pump is 600 liters/hour. The water flow rate value 64 m^3 / hour is still far from the optimal water flow rate reference value 120 m^3/hour so that the decrease in water conductivity is not too large and there has not been optimal contact between the resin and water. Data of up and down conductivity values indicate that three minutes is not enough to reach a stable condition of ion exchange between water and cation resin.

The graph in **FIGURE 3** shows that the greater the water flow rate the smaller the conductivity of the water after passing through the anion resin. Based on the data sheet tulsion resin A23, the optimal water flow rate for ion exchange occurs is 60 m^3/hour. The value of water flow rate passing through resin anion and the conductivity of the water produced is shown in **TABLE 2**.

A decrease in conductivity is evident, starting from 14.6 µS to 5.83 µS. It is caused large flow water that passes through the anion resin not too far from the optimal reference flow water. It results in optimal contact between the resin and water which causes optimal ion exchange as well. The tendency of the decreasing conductivity value indicates that three minutes is sufficient to reach a stable condition of ion exchange between water and anion resin. Water flow rate 20 m^3/hour results in high conductivity because water is not distributed the entire resin, so resin mixing is not optimal.

TABLE 2. Water flow rate passing through anion resin and conductivity of water produced after passing it

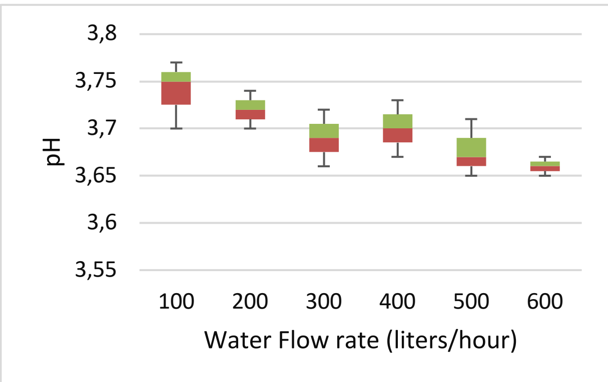
No	water flow rate passing through anion resin (m^3/hour)	the conductivity of the water produced (µS)	Standard Deviation
1	20	14.60	0.80
2	32	10.43	0.60
3	44	8.60	0.56
4	53	6.97	0.42
5	58	6.07	0.42
6	64	5.83	0.40

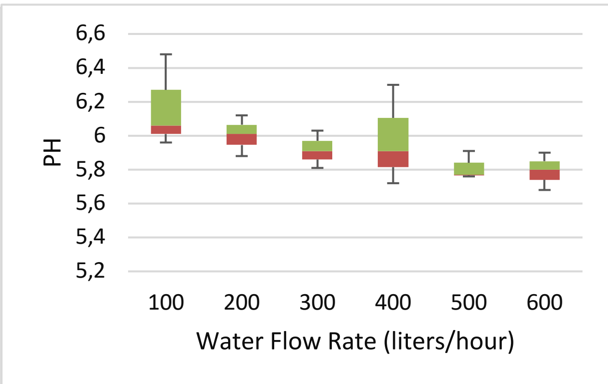
The experiment also showed that the conductivity value after passing through anion resin was smaller than the conductivity value after passing cation resin. It is caused by water that after passing through anion resin occurs twice ion exchange, which is ion exchange in cation and anion resins. The experiment also showed the optimal water flow after passing through the pump for ion exchange to occur is 600 liters/hour, where it produces a water flow rate of 64 m^3/hour after passing through anion resin. But water flow rate 600 liters/hour is too high which can cause the pipe to leak and even break. The standard water flow rate so that the pipe does not leak is 300 liters/hour. However, at the current altitude position of 1.25 m, the water flow rate 300 liters/hour cannot reach the water flow rate after passing through the resin of 64 m^3 / hour. To reach this rate, the position of the water level after passing the pump (h1) 300 liters/hour or conductivity meter for cation and anion resin (h2) must be changed. To find the position of the water level height can be sought from the decline in Bernoulli's law, as follows:

$$h_1 = \frac{V_2{}^2 - V_1{}^2}{2g} - \frac{3P_1}{4\rho g} + h_2 \tag{4}$$

$$h_2 = -\frac{V_2{}^2 - V_1{}^2}{2g} + \frac{3P_1}{4\rho g} + h_1 \tag{5}$$

From the calculation in equation no.4, to get the water flow rate after passing through the resin of 64.3 m^3 / hour, height position h2 is not changed so the height position h1 is 6.7 meters. The acquisition of a water flow rate of 64.3 m^3 / hour can also be got by not changing the position of h1, but the position of water passing through the resin (h2) is changed to -3.8 m. Both of these are very inefficient in water treatment system design because the position of h1 is too high or the position of h2 is too deep. A possible way to obtain a water rate of 64,3 m^3 / hour in the h2 position is a design engineering





(a) (b)

FIGURE 4. (a) Graphic relationship between water flow rate after through resin tank and pH water after through cation resin andm(b) Graphic relationship between water flow rate after through resin tank and pH water after through anion resin

The pH of water in the ion exchange process is closely related to conductivity. The graph in **FIGURE 4** shows that the pH of the water after passing through the anion resin is more alkaline than the pH of the water after passing through the cation resin. This is due to the process of releasing H^+ after passing cation resin and the process of releasing OH^- after passing anion resin. A good cation exchange process can be seen from the large decrease/increase from initial pH to acid pH. A good anion exchange process can be seen from the large decrease/increase from initial pH to alkaline pH. Good conductivity can be seen from the large decrease/increase in pH. On the graph, the water flow rate of 600 liters/hour is the most optimal rate of water exchange of ions. This can be seen from the pH of the most acidic water due to the amount of H^+ release.

Similar research has been done by Widarti S [8]. His research was themed "Effect of Water Rate on the Efficiency of Commercial Cation Exchange Resin Columns and Different Loads of Metal Ion Adsorption". The research explains the number of absorbed metal ions Mg^{2+} and Zn^{2+} decreases with increasing flow rate while Na^+ ions experience the opposite. This is because metal ions with +2 charges such as Mg^{2+} and Zn^{2+} metal ions require two negative functional groups in the resin to neutralize the charge. The position of the negative functional groups in the resin is not necessarily regular and close together or at a distance that is still possible to interact electrostatically with the positive charge of the metal ion so that it takes longer for the Mg^{2+} and Zn^{2+} ions to meet the two functional groups. Therefore, the lower the water flow rate, the more Mg^{2+} and Zn^{2+} ions are adsorbed [8]. The difference this research between the research's author is the object of the research. The research object of the author is the effect of the water flow rate on the ion exchange process. The parameters observed were water flow rate, conductivity, and pH. At the end of the analysis of the research, the author needs to do an engineering water treatment system so that the ion exchange process runs optimally.

CONCLUSION

The amount of water that passes through the resin is very influential on ion exchange. This research shows if the amount of water flow rate pass resin is closer to the value recommended by the datasheet, it can cause the ion exchange process to run optimally. The optimal water flow rate after passing through the pump for the ion exchange process is 600 liters/hour. This flow rate causes water flow rate that passes through cation and anion resin 64 m^3 / hour. However, a water flow rate of 600 liters/hour can cause the pipes to leak and burst. The standard water flow rate passed by the pipe is 300 liters/hour. To overcome this situation, design engineering is needed, so that the pump can be operated at the recommended speed.

ACKNOWLEDGMENT

Thank you for all staff of the Irradiation, Electromechanic, and Instrumentation Division

REFERENCES

[1] D.Reni, M. Munas, N.Jeni, S.Nanda, *Kombinasi Proses Filtrasi dan Ion Exchange Secara Kontinyu ada Pembuatan Aquadm (Demineralized Water)*, Chemica, Vol 4, 27-32(2017).

[2] (D Reni, M Munas, N Jeni, S Nanda, *The combination of continuous filtration and ion exchange processes in the manufacture of aquadm (Demineralized Water)*, Chemica, Vol 4, 27-32(2017).)

[3] R.Aghoyeh, H.Khalafi, *Design of Dual Column Water Purification System for Industrial Gamma Irradiator Based on PUROLITE Resins*, Journal Annals of Nuclear Energy Series 69, 90-96 (2014).

[4] D.Oyekunle and J.Omoleye., *Effect of Particle Sizes on Kinetics of Demineralization of Snail Shell of Chitin Synthesis Using Acetic Acid*, Journal of Heliyon, Vol.5 (2019).

[5] M.Marx et al., *Manufacturing of Demineralized whey concentrates with extended shelf life: Impact of Degree of Demineralization on Functional and Microbal Quality Criteria*, Journal Food and Bioproduct Processing, 1-11, Vol.114 (2019).

[6] R. Arif, S.Bimo, *Karakteristik Kinerja Resin Penukar Ion Pada Water Treatment System Iradiator Gamma Merah Putih*, , Prosiding Seminar Keselamatan Nuklir, 419-425(2019).

[7] (R. Arif, S. Bimo, *Performance Characteristics of Ion Exchange Resins In The Water Treatment System Of Merah-Putih Gamma Irradiator*, National Proceeding of Nuclear Safety Seminar, 419-425(2019).)

[8] N.S Natheer, *Experimental Study on Ion Exchange Rate of Calcium Hardness in Water Softening Process Using Strong Acid Resin Dowex HCR S/S*, College of Engineering Journal (NUJEC),107-114(2016).

[9] R.R Abdul, K.S Ni, R Basuki, *Optimalisasi Hidrolisis Sukrosa Menggunakan Resin Penukar Kation Tipe Sulfonat*, Jurnal Natural Science,119-131(2012).

[10] (R.R Abdul, K.S Ni, R Basuki, *Optimization of Sucrose Hydrolysis Using Sulfonate Type Cation Exchange Resin, Jurnal Natural Science*,119-131(2012).)

[11] Widarti.S, *The Influenced Of Fleed Flow Rate Toward Capacity Of Commercial Cation Exchanger Resin And Adsorption Of Meta Ion With Different Valence*, Sigma MU, Vol 7(2015).

Effect of Accelerator Room Temperature on Vacuum Process Electron Beam Machine GJ-2

Arif Rachmanto [1, a)] and Adam Tirta Kusuma [1, b)]

[1] *The Center for Isotope and Radiation Application, Lebak Bulus Street No.49 South Jakarta*

[a)] Corresponding author: arif_rachmanto@batan.go.id
[b)] Another author a.tirta-kusuma@batan.go.id

Abstract. Vacuuming is the initial step and important process in the operation of the electron beam machine. Vacuuming reduces collisions between electrons and air particles in the accelerator tube to the scanning horn so that irradiation runs optimally. In the electron beam machine GJ-2, there is a vacuum interlock system that serves as a safety for the turbo pump. The turbo pump can be operated if the vacuum pressure reaches 1.49 x 10^{-2} torr. When the accelerator room temperature value is set at 24ºC - 26ºC, the vacuum process is difficult to reach 1.49 x 10^{-2} torr. Therefore, this study needs to be done to obtain the standard value of the accelerator room temperature so that pressure 1.49 x 10^{-2} torr is reached during the vacuum process. The method used in this study is to analyze the vacuum pressure data with accelerator room temperature variations. The data are taken when the machine has not operated for two days. The results show that the lower room temperature, the easier vacuum process. The standard value of the accelerator room temperature to pressure reached 1.49 x 10^{-2} torr is 21ºC. When the electron beam machine has not been operated, there is an increase in vacuum pressure is caused by gas permeation and outgassing.

Keywords: temperature, vacuum, electron beam machine, pressure

INTRODUCTION

The electromagnetic spectrum is made up of both ionizing and non-ionizing radiation frequencies [1]. Food irradiation is an irradiation technique using ionizing radiation so that it depends on energy [2]. The primary difference between ionizing and non-ionizing radiation is based on their respective energies as to whether they can ionize the atoms they come into contact with irradiated material [3]. One ionizing irradiation using electron is used for polymerization, sterilization, food safety, etc. The Electron beam machine GJ-2 or MBE GJ-2 is a machine used to irradiate a product using an electron beam source. The electrons used to come from the tungsten element, heated with a certain electric current so that electron is emitted and released from its bonds. Electrons are accelerated in the accelerator tube and shot at a product or sample. Electron beam machine GJ-2 has an electron energy 2 MeV with a beam current of 10 mA [4]

In an electron beam machine, there is one main system that must be present in the beam machine itself, which is the vacuum system on the accelerator tube. A vacuum system is a system in which the vacuum of air is carried out in the accelerator tube. The purpose of vacuuming in an electron beam machine is to make the charged particles move freely without obstacles and not collide with air particles. The operation of the turbo pump electron beam machine GJ-2 can be carried out if the vacuum pressure reaches 1.49 x 10^{-2} torr. To achieve this pressure, rotary pumps are operated within a certain period. Also, there is another factor that influences the achievement of that pressure, i.e accelerator room temperature.

Overheated accelerator room temperature causes the vacuum process can not reach pressure 1.49 x 10^{-2} torr. The accelerator room temperature must be adjusted so that the vacuum process can run well. Therefore, this study needs to be done to obtain the standard value of the accelerator room temperature so that pressure 1.49 x 10^{-2} torr is reached during the vacuum process

BASIC THEORY

Electron Beam Machine GJ-2

The electron beam machine GJ-2 is a machine used for irradiation using an electron beam source. Electron beam machine GJ-2 is one of the irradiation facilities owned by the Center for Isotope and Radiation Application, BATAN. This facility was established in 1993 and inaugurated by President Soeharto in 1994. The technical specifications of the electron beam machine GJ-2 are as follows:

Name	: MBE GJ-2 (GJ=Gao Jia: *high voltage accelerator*).
Type	: Dynamitron, with a mixture of CO_2 and N_2 gas isolators.
Electron energy	: 0.7-2.0 MeV, can be set continuously
Current beam	: 0-10 mA, can be set continuously
Scanner width	: 80-120 cm, can be set continuously
Maximum power	: 20 kW
Operation	: Power below 20 KW, operating duration above 8 hours [5]

By having very high energy, the electron beam machine has a wide scope in its utilization. The electron beam machine GJ-2 carries out irradiation activities for pasteurization, sterilization, degradation, vulcanization, grafting, and cross-linking. The electron beam machine GJ-2 is included in the category II electron beam machine which is placed in the shielding room and cannot be accessed while operating using an entry control system [6]. **FIGURE 1** shows the design of the category II electron beam machine.

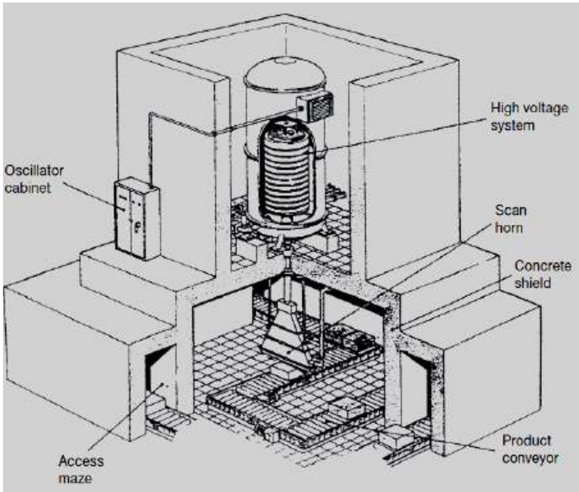


FIGURE 1. Category II Electron Beam Machine Design [5]

FIGURE 1 shows the design of the electron beam machine. The electron beam machine GJ-2 consists of several main components:

1. Source: Electrons originating from tungsten that are electrified by certain electric currents.
2. Accelerating system: To accelerate electrons so they can reach the desired energy.
3. Vacuum system: To vacuum the accelerator tube so that the electrons can move freely
4. Focusing system (optical): For directing the electron by its path.
5. Conveyor system: To bring the product/sample into the irradiation room.
6. Control panel system: To control all existing systems on the electron beam machine

The Vacuum System for Accelerator Tube

The vacuum system has an important role in the product irradiation process. The accelerator tube as in Figure 2 must be in a vacuum condition where it has a very low gas density so that electrons can move freely.

The vacuum value used in the electron beam machine vacuum system is in the order of 10^{-6} torr. This value is a condition that must be met before raising the high voltage and outputting a beam current [5]. Before reaching the vacuum value in the order of 10^{-6} torr, several stages that must be done. First, the rotary pump must be activated to make the initial vacuum until 1.49 x 10^{-2} torr. Second, after the vacuum value is reached the turbo pump is activated to reach the vacuum value in order 10^{-6} torr. In each stage of the vacuum process, there is an interlock mechanism that is used for safety.

The word vacuum comes from the Latin "vacua" which means empty. In a vacuum, some air and other gases are removed from the chamber which a volume filled with air [8]. Thus, a vacuum is a condition of the room where some of the air and other gases have been evacuated so that the pressure is below atmospheric pressure [9]. The vacuum range according to John F. O'Hanlon is divided into 6: low vacuum, medium vacuum, high vacuum, very high vacuum, ultra high vacuum, and extra high vacuum. The vacuum level ranges are shown in the table below.

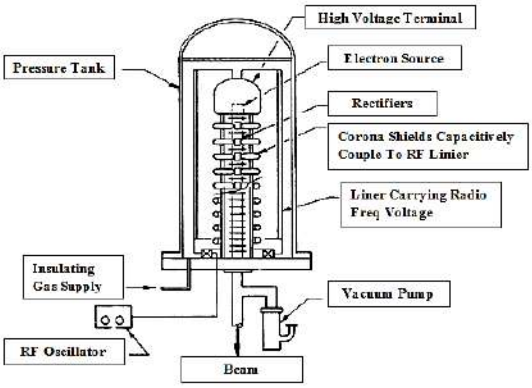


FIGURE 2. Schematic of Accelerator Tube for Dynamitron Type [7]

Vacuum

TABLE 1. Range of Vacuum Levels According to John F. O'Hanlon [9]

No	Vacuum Level	Vacuum Range (Pa)	Vacuum Range (Torr)
1	Low	$10^5 > P > 3.3 \times 10^3$	$7.5 \times 10^2 > P > 24.75$
2	Medium	$3.3\times 10^3 \geq P \geq 10^{-1}$	$24.75 \geq P \geq 7.5 \times 10^{-4}$
3	High	$10^{-1} \geq P \geq 10^{-4}$	$7.5 \times 10^{-4} \geq P \geq 7.5 \times 10^{-7}$
4	Very High	$10^{-4} \geq P \geq 10^{-7}$	$7.5 \times 10^{-7} \geq P \geq 7.5 \times 10^{-10}$
5	Ultra High	$10^{-7} \geq P \geq 10^{-10}$	$7.5 \times 10^{-10} \geq P \geq 7.5 \times 10^{-13}$
6	Extra Ultra High	$P \leq 10^{-10}$	$P \leq 7.5 \times 10^{-13}$

METHODS

Materials and Tools

The Materials and tools used in this study are the electron beam machine GJ-2, hygrometer, thermometer, timer, and valve gauge meter. The electron beam machine is equipped with a rotary pump and turbo pump. The rotary pump is used to vacuum the accelerator tube to the pressure of 10^{-3} torr. The turbo pump is used to vacuum the accelerator tube to a pressure of 10^{-7} torr. A thermometer is used to measure accelerator room temperature. A timer is used to set and determine the vacuum time and the valve gauge meter is used to display the vacuum pressure on the accelerator tube.

Work method

The steps used to analyze the effect of temperature on the electron beam machine GJ-2 vacuum process are as follows: variations room temperature must be determined. The temperature values taken were 26°C, 22°C, 21°C, 20°C and 19°C. The rotary pump is turned on until certain pressure. Time and vacuum pressure are recorded. Data in the graph form of vacuum pressure with a vacuum time. Collection data of vacuum is taken when the electron beam machine is not operated by vacuum for two days

RESULT AND DISCUSSION

The highest pressure reduction in the vacuum process of 26oC was found in the first 60 minutes. The reduction in vacuum pressure is not significant after 60 minutes. Pressure toward the constant value after vacuuming for 447 minutes with a pressure 0.0207 torr. It has not to vacuum pressured 1.49 x 10-2 torr so that turbo pump can not be operated. The fastest vacuum pressure reduction at the beginning of the vacuum because the number of air molecules is much. A large number of molecules will be easier to be evacuated compared to a small number of molecules. So, the reduction in vacuum pressure will be faster.

The data in **FIGURE 4** is taken when the electron beam machine is not operated by vacuum for two days. The machine has not operated since the vacuuming at a temperature of 26°C. The highest pressure reduction in the vacuum process of 22°C was found in the first 60 minutes. After that, it occurs a reduction not significantly. Pressure toward the constant value after vacuuming for 445 minutes with a pressure 0.0174 torr. It has not still vacuum pressured 1.49×10^{-2} torr so that turbo pump can not be operated.

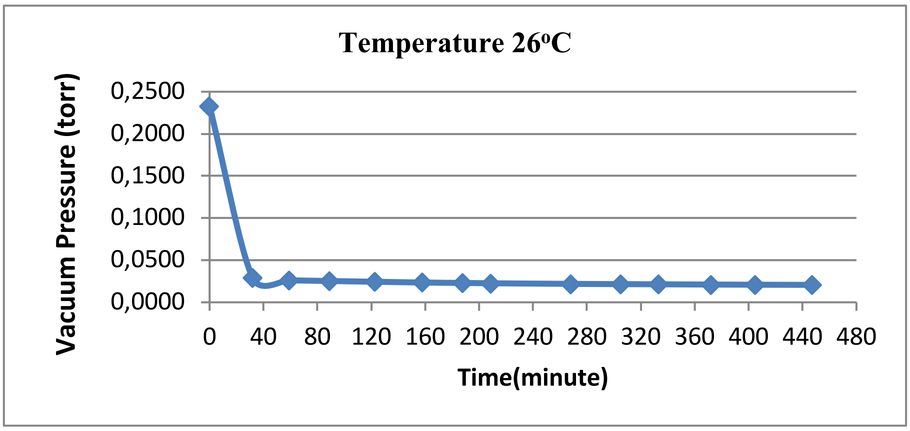


FIGURE 3. Graph of the Relationship between Temperature and Time to the process of vacuuming the Electron beam machine GJ-2 at 26°C.

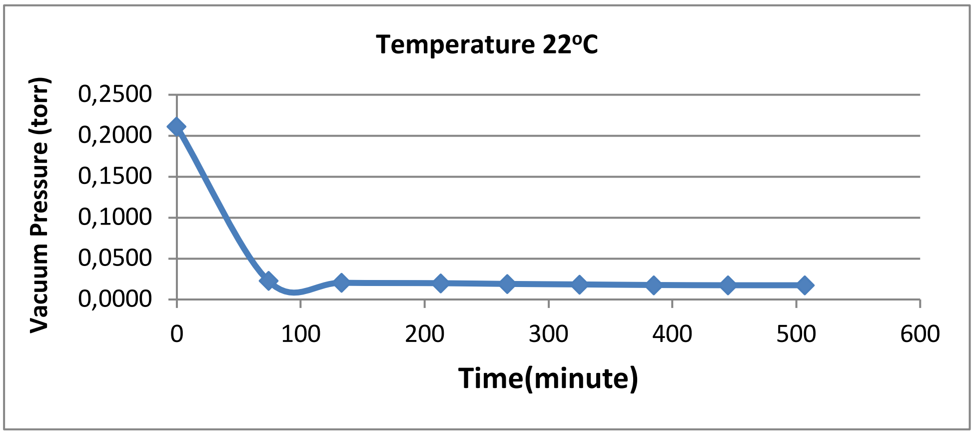


FIGURE 4. Graph of the Relationship between Temperature and Time to the process of vacuuming the Electron beam machine GJ-2 at 22°C.

The trend graph in **FIGURE 5** is similar to the graph in **FIGURES 3** and **4**. The data in **FIGURE 5** is taken when the electron beam machine is not operated by vacuum for two days. The machine has not operated since the vacuuming at a temperature of 22°C. The highest pressure reduction in the vacuum process of 21°C was found in the first 70 minutes. After that, it occurs a reduction not significantly. Pressure values start constantly after vacuuming for 420 minutes with pressure 0.0149 torr. It has reached pressure 1.49 x 10^{-2} torr so that turbo pump can be operated.

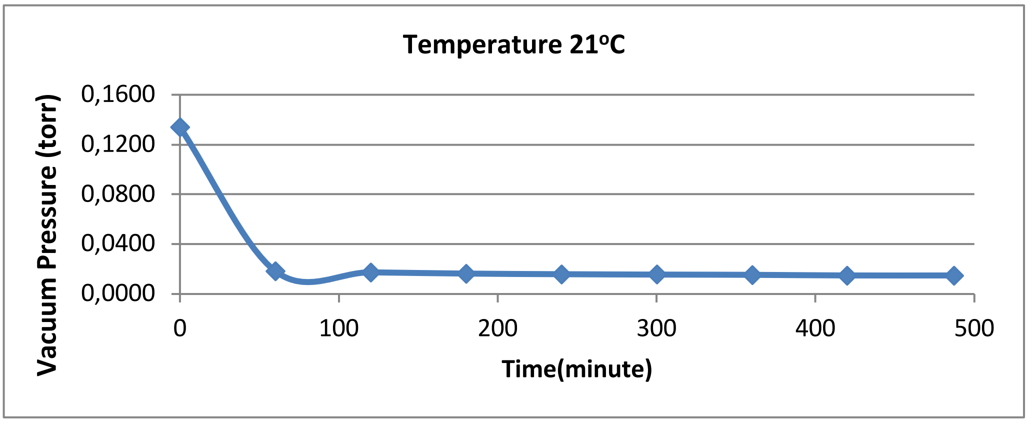


FIGURE 5. Graph of the Relationship between Temperature and Time to the process of vacuuming the Electron beam machine GJ-2 at 21°C.

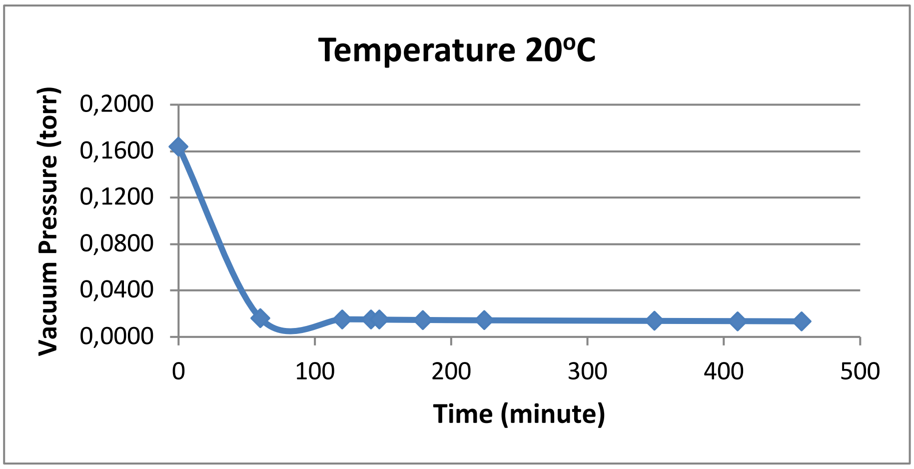


FIGURE 6. Graph of the Relationship between Temperature and Time to the process of vacuuming the Electron beam machine GJ-2 at 20ºC.

The data in **FIGURE 6** is taken when the electron beam machine is not operated by vacuum for two days. The machine has not operated since the vacuuming at a temperature of 21ºC. The trend graph in **FIGURE 6** is similar to the graph before. There is the highest reduction of vacuum pressure in the first 60 minutes. The time required to reach a pressure of 1.49 x 10^{-2} torr is 89 minutes. Pressure continues to reduce until 1.34 x 10^{-2} torr and the time needed to reach it is 457 minutes. The pressure achieved at temperature 20ºC is lower than temperature 21ºC. The pressure achieved has exceeded 1.49 x 10^{-2} torr, Of course, turbo pump can be operated.

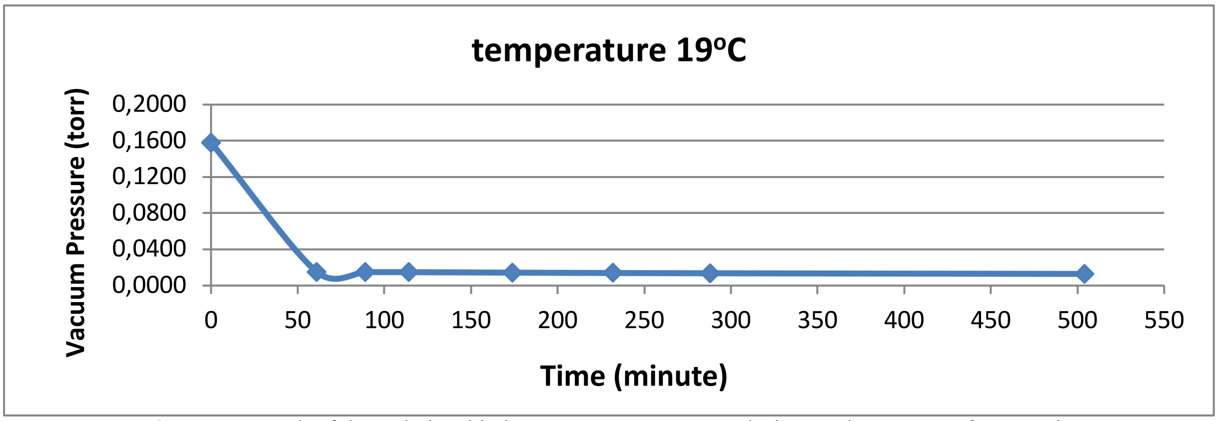


FIGURE 7. Graph of the Relationship between Temperature and Time to the process of vacuuming the Electron beam machine GJ-2 at 19oC.

The data in **FIGURE 7** is taken when the electron beam machine is not operated by vacuum for two days. The machine has not operated since the vacuuming at a temperature of 20ºC. There is the highest reduction of vacuum pressure in the first 61 minutes. Pressure continues to reduce until 1.29 x 10^{-2} torr and the time needed to reach it is 504 minutes. All five experiments show is the lower the temperature the faster the vacuum process. One of the influencing factors is the effect of temperature on the gas law. The gas law that applies to this vacuum is the ideal gas law. This condition applies to the ideal gas law because the volume of a molecule is negligible to the volume of space occupied and the attraction between molecules is so small that it can be ignored. This ideal gas has the following equation:

$$PV = nRT \ldots\ldots\ldots\ldots\ldots (1)$$

Where P = gas pressure on the accelerator tube (N/m^2)
V = Gas volume on the accelerator tube (m^3)
n = number of particles
R = universal gas constant (R= 8,315 J/mol.K)
T = Temperature (K)

From the ideal gas equation, it can be seen that the pressure and temperature are directly proportional. The lower the temperature, the lower the gas pressure so that vacuum at low temperatures will be faster. This is the cause at low temperatures, vacuum value is more easily achieved than at high temperatures.

If looked at the graph, there is a rise in pressure when the electron beam machine has not operated for two days. The following is a table of data rise pressure when the machine has not operated for two days:

TABLE 2. Data rise pressure when the machine has not operated for two days

No	Initial Pressure (torr)	Initial Temperature	Pressure after two days (torr)	Last Temperature	Pressure rise (torr)
1	0.0207	26°C	0.2110	22°C	0.1903
2	0.0174	22°C	0.1340	21°C	0.1166
3	0.0149	21°C	0.1640	20°C	0.1491
4	0.0134	20°C	0.1580	19°C	0.1446

The increase in vacuum pressure can be caused by several things, including leakage, evaporation, permeation, and outgassing. Leakage testing is carried out by not operating the vacuum for two weeks. Vacuum pressure during the last operation was 0.0147 torr. After not being operated for two weeks, the pressure becomes 0.240 torr. One indication of leakage is after a long time it has not been operated, the vacuum pressure will rise > 1 torr. So, the leakage factor is not the cause of the increase in vacuum pressure in this case.

Evaporation is caused by the presence of materials in the system because of surface uncleanness or volatile matter at low pressure. The evaporation factor is also not a cause of increased vacuum pressure because the components in the accelerator tube have been cleaned when overhaul. So, the factors that cause an increase in vacuum pressure are gas permeation and outgassing.

Gas permeation and outgassing include dynamic loads [10]. Permeation is the entry process of gas molecules/ atoms from the outer surface which have atmospheric pressure to the inner surface which has a lower pressure because it is vacated. Gas outgassing (release of gas) is caused by diffusion and desorption. Gas diffusion occurs when a gas molecule/atom attaches to the surface which then enters the surface of the wall, after vacuuming the gas molecule / atom out and off the surface [11]. So, the material used in vacuum systems especially ultra high vacuum must have low outgassing when working at high temperatures and the electron gun operating [12]

CONCLUSION

The results of this study indicate the temperature limit of the accelerator room to reach a vacuum pressure of 1.49×10^{-2} torr is 21°C. It is used as a minimum standard vacuum pressure so that the turbo pump can be operated. When the vacuum is not operated for some time there is an increase in the value of the vacuum. Factors that cause an increase in vacuum pressure are permeation and gas outgassing.

ACKNOWLEDGMENT

Thank you for all staff of the Irradiator and Electron Beam Division.

REFERENCES

[1] D.P Surresh and S. Shima, "Electron Beam Technology and Other Irradiation Technology Applications in Food Technology", Springer International Publishing Switzerland, (2016), pp.249-267

[2] Miller R.B, *Electronic irradiation of foods: an introduction to the technology* (Springer, New York,2005) pp. 350.

[3] Chmielewski A.G, Kang C.M, Kang C.S, and Vujic J.L, *Radiation technology: Introduction to industrial and environmental applications.* (Seoul National University Press, Seoul,2006), pp. 274.

[4] Shanghai Xian Feng Electric Manufacture Works, *Technical Documentation of Operating and Maintenace Manual for Accelerator Model GJ-2* (China,1993)

[5] M, Nada, "*Sistim Interlock Mesin Berkas Elektron GJ-2"*, Prosiding Pertemuan dan Presentasi Ilmiah Teknologi Akselerator dan Aplikasinya, (1999), pp.34-42, vol.1, no. 1

[6] (M, Nada, "*Interlock System of Electron Beam Machine GJ-2*", Proceeding of Meeting and Scientific Presentations on Accelerator Technology and Its Applications, (1999), pp.34-42, vol.1, no. 1)

[7] IAEA , *Radiation Safety of Gamma Electron and X-Ray Irradiation Facilities* (2010)

[8] W E Gundaker, F B Richard, *Recomendation for the Safe Operation of Particle Accelerator* (1968)
[9] M. Kaminsky, J.M Lafferty, *Dictionary For Vacuum Science and Technology* : American *Vacuum Society, NewYork* (1980)
[10] Suprapto, W. Susilo, *Pengenalan Teknologi Vakum* (Pustaka Pelajar, Yogyakarta, 2017)
[11] Suprapto, W. Susilo, *Introduction of Vacuum Technology* (Pustaka Pelajar, Yogyakarta, 2017))
[12] Suhartono, Sukidi, "*Pengujian Sistem Vakum Mbe 350KeV/10 Ma Pasca Penggantian Pompa Turbomoleku*l", Prosiding Penelitian dan Pengelolaan Perangkat Nuklir, (2011), pp.203-207.
[13] (Suhartono, Sukidi, "*Testing of Mbe 350 KeV / 10 Ma Vacuum System after Turbomolecular Pump Replacement*", Proceedings of Research and Management of Nuclear Devices, (2011), pp.203-207)
[14] Darsono, Suprapto, S. Rany "*Analysis dan Pengujian Komponen Tabung Sistem Vakum Mesin Berkas Elektron 350 keV/10 mA*, Prosiding dan Pertemuan Presentasi Ilmiah Teknologi Akselerator dan Aplikasinya,(2003), Vol.1, pp.1-10
[15] (Darsono, Suprapto, S. Rany, "Analysis and Testing of 350 keV/10 mA Electron Beam Machine Vacuum System Tube Components, Proceeding and Meeting of Scientific Presentations on Accelerator Technology and Its Applications, ,(2003), Vol.1, pp.1-10)
[16] Pandiyar M L, 2008, "*Ultra High Vacuum Test Setup for Electron Gun, Journal of Physics*" Conference Series 114(2008) 012601.

Calibration of LiF Type Personal Dosimeter Against Mixed Beta-Gamma Radiation

Rofiq Syaifudin[1, a)], Egnes Ekaranti[1], Sri. S. Lolaningrum[1], and Weni Purwanti[2]

[1]*Center for Technology Radiation Safety and Metrology (PTKMR BATAN)*
Jl. Lebak Bulus Raya No. 49 Jakarta Selatan
[2]*Brawijaya University, Malang East Java*

a) rofiq@batan.go.id

Abstract. Widely utilization of radiation sources as a means of measuring and testing demanded the monitoring for the safety of radiation workers intensified. Different types of radiation that are applied are the photon (gamma / X-rays), neutrons, beta, and radiation mixed. One of the most commonly used radiation monitors is a personal dosimeter. The dosimeter must be calibrated before using a radiation monitor. The calibration to radiation mixed of beta-gamma has been done by exposing a dosimeter with a gamma source of ^{137}Cs, and beta (^{90}Sr, ^{85}Kr, and ^{147}Pm) following required standards. Calibration was done with a single source before dose variation that has been determined then be evaluated to obtain dose-response curves. Furthermore, the irradiation mixture was done by a combination of beta-gamma dose of 1 mSv of beta + 3 mSv of gamma and vice versa. The result was linear dose-response curves with a correlation coefficient of 0.99 for all of the radiation sources were obtained. While the exposure in radiation mixture was relatively calculation dose obtained close to the transfer doses with a deviation below 20 %.

Keywords: calibration curve, personal dosimeter, mixed field radiation, beta-gamma

INTRODUCTION

There are many places where mixed field radiation can be found, among of them are a special place in a nuclear power plant, as well as other activities related to the nuclear fuel cycle, can also be found in medical activities and high energy accelerators research, at the high altitude of the civil and military aviation and in space exploration activities.

Mixed field radiation consists of various types of radiation and/or energy, such as photons and electrons, photons and neutrons, or even neutrons with considerable energy difference can also spread some of the characteristics mixed field. The mixed field can also be composed of a mixture of different natural radiation but with the same weight factors, eg. beta-photon field [1]. Thus, the type of radiation used to calibrate dosimeters are photons, neutrons, and beta particles [2]. As a basic quantity and calibration method described following ISO [3,4,5,6] was a unit that has been determined by the primary standard laboratory, for example; fluence of the neutron, exposure of air Kerma for photons, and absorbed dose for beta radiation.

Calibration can be defined as an operation or work done on the condition that has been determined to establish a relationship between the values given by the measuring instrument or the system following the true value of the quantity measured. The paper will be presented the calibration of personal dosimeters made of LiF material against mixed beta-gamma radiation. The Gamma source used was 137Cs while the beta source was 90Sr, 85Kr, and 147Pm.

Given the experimental measurements of the operational equivalent dose quantity of mixed field, some practical problems arise because it is usually difficult to measure a dose equivalent with a single detector. This difficulty is caused by a different sensitivity, the application of different calibration factors for each component field, or with different measurement conditions, such as in the measurement of radiation penetrating and non-penetrating. In the mixed field is generally more a rule than an exception, though in practice the doses caused by one of the component fields were larger then the contributions of the others can be ignored.

PROCEDURE

Thermoluminescence Dosimeter (TLD) 7776 with Holder 8814 has been used in the calibration. The TLD is made from LiF with ^{7}LiF enrichment to respond to beta-gamma mixed radiation. The holder of TLD has a variety of filters that can be converted at various depths to facilitate the monitoring of radiation received by skin, eye lens, and at a depth of 10 mm from the surface of the skin. TLD reading device is reader Model 6600 Harshaw. Beta sources used to calibrate were ^{90}Sr, ^{85}Kr, and ^{147}Pm while the gamma source was ^{137}Cs.

The beginning step was annealing to eliminate the remnants of the radiation /electron that may still exist in the electron trap. The next step was a dose uniformity test of approximately 20 TLDs with a single dose of 1 mSv gamma, then uniform dose responses are grouped with a deviation of about 5 % [7]. The dose uniformity test was also conducted to the source of ^{90}Sr (absorbed dose rate of 252.79 mGy/h), ^{85}Kr (absorbed dose rate of 93.71 mGy/h), and ^{147}Pm (absorbed dose rate 0.06 mGy/h) by using Beta Secondary Standard type 1 (BSS 1) static standard. In the case of absorbed dose rate (mGy/h), the radiation quality can be taken as equal to unity for external radiation [8]. So that the unity mGy/h is as similar to mSv/h.

Irradiation to uniformity of response for beta using 12 pieces TLD selected from uniformity test against gamma source. Each 4 pieces TLD arranged in phantom and irradiated with ^{90}Sr source at a dose of 2.12 mSv (30 seconds), the source of ^{85}Kr at a dose of 0.78 mSv (30 seconds), and ^{147}Pm source at a dose of 0.1 mSv (1 hour and 36 minutes).

FIGURE 1. Exposing TLD against beta source (^{90}Sr, ^{85}Kr, and ^{147}Pm) with using BSS 1

$$H = DQ \tag{1}$$

Where H is the equivalent dose (Sv), D is absorbed dose (Gy), and Q is the radiation quality factor (dimensionless), for beta, X-ray, and gamma, Q can be taken as equal to unity for external radiation [8].

FIGURE 2. Exposing TLD against gamma source ^{137}Cs of OB 85

The next step was exposed to TLD against the 137Cs source. TLD was attached to the surface of polymethylmethacrylate (PMMA) size of 30x30x15 cm facing the source at a distance of 200 cm from the focal spot of the source (SDD). Irradiation on a standard field dose of 0.1 mSv to 200 mSv is done [9]. Calibration is performed only in one direction (0 °) between the source to the dosimeter.

TLD was saved for about 24 ± 3 hours and then read the response and analyzed the results of calibration. Calibration of the three sources of beta is also done in the same way as the time of irradiation to a dose uniformity test. The beta source placed on the stand calibration standard (BSS-1) within the distance of exposure was 30 cm for ^{90}Sr and ^{85}Kr, while for ^{147}Pm was 20 cm long. However, there are differences in dose variation given for adjusting the dose rate at any source of beta and exposing the time duration that can be given. From the calibration performed on four sources of radiation was created curve relationship between exposure doses against the response of TLD.

The final step was exposed to a mix of beta-gamma radiation. Exposure applied for mixed of ^{90}Sr0 + ^{137}Cs, ^{85}Kr + ^{137}Cs, and ^{147}Pm + ^{137}Cs with variation dose given of 1 mSv + 3 mSv (beta + gamma) and vice versa. The result of exposure of radiation mixture was analyzed by using a dose-response curve that has been obtained previously for calculating the radiation dose received TLD, named measured dose (D_M), and compared to the exposed dose given, named actual dose (D_T). To determined exposure time for the desired dose could be used Equation (2).

$$Time\ (hr) = \frac{Desired\ dose\ (mSv)}{Dose\ rate\ (\frac{mSv}{hr})} \tag{2}$$

RESULT AND DISCUSSION

In response uniformity test dose for 20 TLDs gained of 12 TLD uniform with the deviation for about 5 %. The 12 pieces of TLD were then used for calibrating of dosimeters against beta and gamma sources that have been determined.

TLD calibration of the gamma source is shown in **FIGURE 3.** The radiation dose given were 0.1; 0.3; 0.5; 1; 1.5; 3, and 5 mSv. From the figure, it can be seen that the dose and response curves obtained are linear with a correlation coefficient (R) 0.998.

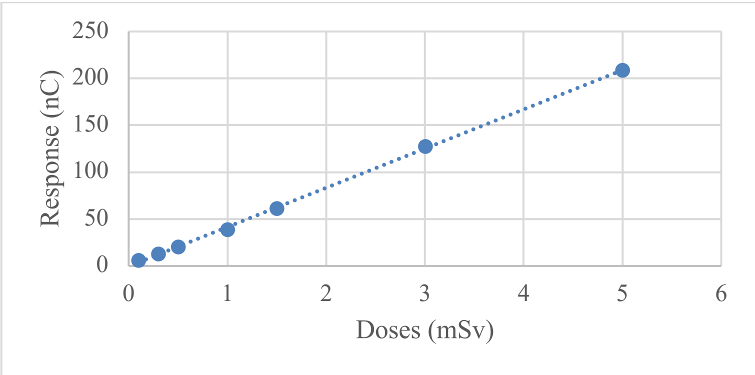


FIGURE 3. Dose calibration curves and TLD response to the ^{137}Cs source

To calibrate dosimeters against sources of ^{90}Sr, the doses variations given were 0.15; 0.35; 0.49; 1.05; 1.54; 3.02; and 5.05. The dose given is the calculation of the absorbed dose rate of the source and time required to produce the equivalent dose. The duration (time) irradiation is used as a criterion in exposures because of the difficulty determining the proper doses. From the time specified, the dose can be known. The relationship curve between the dose given during the calibration and the measured response is shown in **FIGURE 4**. From the figure can be seen that the dosimeter response is linear with a correlation coefficient (R) 0.998.

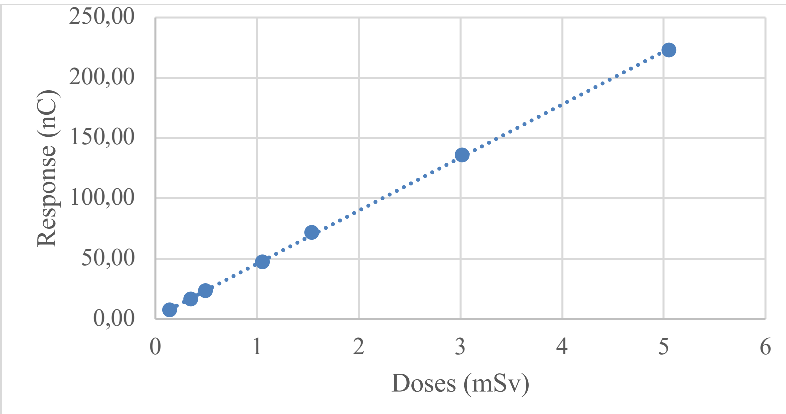


FIGURE 4. The relationship curve between the exposure doses and the measured response of ^{90}Sr

In the calibration of dosimeter against source of ^{85}Kr, doses that can be given were 0.11; 0.31; 0.52; 1.02; 1.51; 3.02; and 5 mSv. This variation doses also unlike slightly the previous source, also encountered difficulties to obtain the equivalent dose appropriate because of measured dose rate is also not in round number. However, the curve obtained from the evaluation showed a linear relationship with a correlation coefficient (R) 0.998.

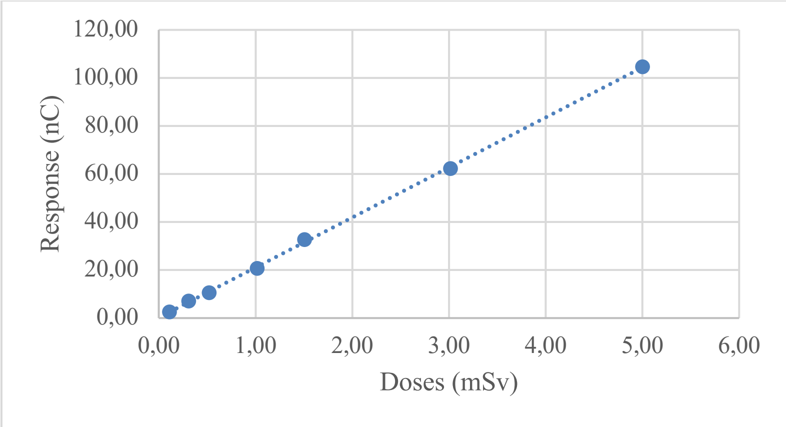


FIGURE 5. The relationship curve between the exposure doses and the measured response of ^{85}Kr

Calibration was rather difficult to ^{147}Pm due to the absorbed dose rate was so low that the exposure time needed for some desired dose variation was too long. From the timing of the irradiation of the obtained dose variation of 0.1; 0.3; 0.5; 1; 1.5; and 3 mSv. A dose of 5 mSv could not present as necessary because of a very long time until more than 60 hours to reach. The calibration curve dose-response relationship dosimeter is presented in **FIGURE 6**. The curves obtained from the evaluation showed a linear relationship with a correlation coefficient (R) 0.996.

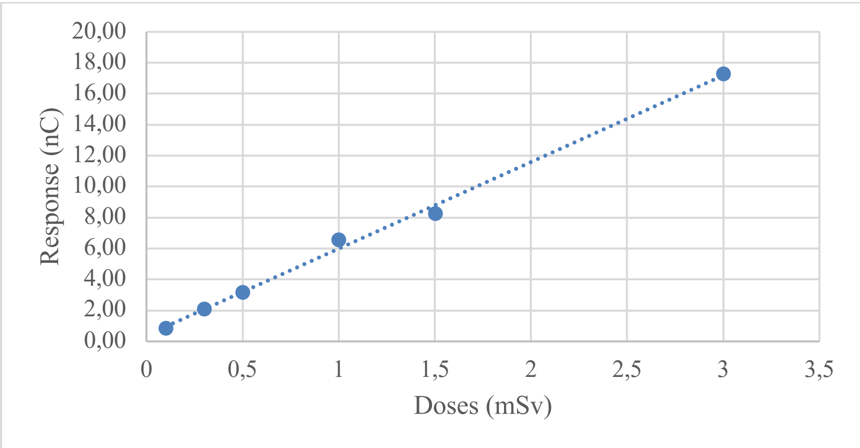


FIGURE 6. The relationship curve between the exposure doses and the measured response of ^{147}Pm

Upon irradiation of mixed beta-gamma radiation, the results obtained are presented in Table 1. From the data presented to the variation of 1 mSv gamma + 3 mSv beta, then for ^{90}Sr and ^{85}Kr, the dosimeters respond to radiation is very well with the deviation to the source of calibration below 10 % while for ^{147}Pm had a significant difference, this was due to a shift of the source seat (BSS-1).

TABLE 1. The result of Mixed Beta-Gamma Radiation

Source (beta + gamma)	**Response of TLD**		**Calculation Dose (mSv)**		**Deviation DM against DT (%)**		**Ratio, DM/DT**	
	Gamma (ii)	**Beta (iii)**	**Gamma**	**Beta**	**Gamma**	**Beta**	**Gamma**	**Beta**
Exposed dose: gamma 1 mSv and beta 3 mSv								
Sr-90 + Cs-137	44,02	169,16	1.06	2,80	5,81	4,20	1,06	0,62
Kr-85 + Cs-137	40,52	104,20	0,98	3,05	13,50	0,46	0,86	0,62
Pm-147 + Cs-137	125,77	143,98	0,97	1,82	94,34	5,31	0,06	0,62
Exposed Dose: gamma 3 mSv and beta 1 mSv								
Sr-90 + Cs-137	119,91	168,48	2,87	1,06	6,81	6,03	0,93	0,54

Kr-85 + Cs-137	125,77	143,98	3,01	0,86	1,58	2,35	1,02	0,53
Pm-147 + Cs-137	118,53	119,22	2,84	0,06	39,35	3,45	0,61	0,53

The comparison of measured doses with actual doses at the ^{90}Sr, ^{85}Kr, and ^{137}Cs sources is still within the upper (UL) and lower limits (LL) set by the IAEA and EURADOS. The comparison of the measured dose with the actual dose at the 147Pm source is beyond the upper and lower limits set by the IAEA and EURADOS, so the results of this comparison are declared to not meet the standard.

The IAEA recommends that one of the criteria for accuracy in measuring the individual dose of gamma is the fulfillment of the ratio factor between the measured dose and the actual dose between 0.67 (-33%) to 1.5 (50%) (IAEA, 1999).

TABLE 2. The test result of Beta-Gamma Radiation that ever been conducted

Source	D_T	D_M	D_M/D_T	EURADOS		IAEA	
				UL	LL	UL	LL
^{137}Cs + ^{90}Sr	3	1,06	0,93	1,60	0,56	2,00	0,54
	1	2,87	1,06	1,54	0,63	2,00	0,62
^{137}Cs + ^{85}Kr	3	0,98	1,02	1,61	0,55	2,00	0,53
	1	3,01	0,86	1,54	0,63	2,00	0,62
^{137}Cs + ^{147}Pm	3	0,97	0,61	1,61	0,55	2,00	0,53
	1	2,84	0,06	1,54	0,63	2,00	0,62

CONCLUSION

TLD used for calibration could respond well. It can be seen from the dose-response relationship curve TLD readings that linear with the correlation coefficient of 0.99 for all sources of radiation used.

In the testing of beta-gamma, mixed radiation results obtained using a dose-response curve relative approaching with actual doses with deviation is less than 20%.

Thus the development of this calibration method can be applied for calculating or evaluating individual radiation dose using this TLD, based on dose-response curves that have been generated.

ACKNOWLEDGMENT

We would like to thank The Sub Division of Calibration, Center for Technology Radiation Safety and Metrology (PTKMR BATAN) for facilitating this research.

REFERENCE

[1] [A. Delgado., Mixed Field Dosimetry**.** 10th International Congress of The International Radiation Protection Association. IRPA-10. Hiroshima 2000.

[2] G. Dietze, Dosimetric Concepts and Calibration of Instruments. Physikalisch-Technische Bundesanstalt, 38116Braunschweig, Germany 2002.

[3] ISO 4037. X and gamma reference radiations for calibrating dosemeters and dose rate meters and for determining their response as a function of photon energy.

[4] ISO 4037-1 (1996) - Part 1: Radiation characteristics and production methods.

[5] ISO 4037-2 (1997) - Part 2: Dosimetry for radiation protection over the energy range 8 keV to 1.3 MeV and 4 MeV to 9 MeV.

[6] ISO 4037-3 (1999) - Part 3: Calibration of area and personal dosemeters and the measurement of their response as a function of energy and angle of incidence. International Organization for Standardization, Geneva, Switzerland

[7] ISO 6980 (1996). Reference beta-radiations for calibrating dosimeters and dose rate meters and for determining their response as a function of beta-radiation energy.

[8] ISO/CD 6980-2 (2000). Beta particle reference radiations - Part 2: Calibration fundamentals related to the basic quantities characterizing the radiation field.
[9] ISO 6980-3 (1998) - Part 3: Calibration of area and personal dosemeters and the determination of their response as a function of beta-radiation energy and angle of incidence. International Organization for Standardization, Geneva, Switzerland
[10] ISO 10647 (1996) Procedures for calibrating and determining the response of neutron-measuring devices used for radiation protection purposes. International Organization for Standardization. Geneva, Switzerland
[11] ISO 6980-1, Nuclear energy — Reference beta-particle radiation - Part 1: Methods of production. 2006
[12] International Commission on Radiological Protection 60:1991, Recommendations of the International Commission on Radiological Protection. Annals of the ICRP, Vol. 21, No. 1 to 3. 1991
[13] International Commission on Radiological Protection. General Principles of Monitoring for Radiation Protection Workers. ICRP 35 (Oxford: Pergamon). 1982.

Radiation Dose of Industrial Radiography Workers and Its Safety Aspects in Indonesia

Egnes Ekaranti[1, a)], Rofiq Syaifudin[1], Irma D. Rahayu[1] and Sri. S. Lolaningrum[1]

[1]*Center for Technology Radiation Safety and Metrology (PTKMR BATAN)*
Jl. Lebak Bulus Raya No. 49 Jakarta Selatan

[a)] egnesekaranti@batan.go.id

Abstract. Various industries applying radiation techniques of non-destructive testing methods using high activity source that has a high risk to the workers. The industrial radiography operating organization must comply with regulations that include the radiation dose of workers. The Radiation dose of all workers of the licensed holder was collected and then accumulated annually. The accumulated dose is classified whether it exceeds the dose limit value or not. The results show the radiation doses were varying. The highest effective dose of radiation worker was 126.79 mSv which is accumulating for three consecutive years. The exceeded dose limit value shows that the radiation safety aspect has not been applied maximally by the industrial radiography operating organization. The safety actions must be conducted by all parties to prevent the adverse health effect related to radiation exposure of the industrial radiography workers.

Keywords: Radiation dose, industrial radiography, NDT, safety aspects

INTRODUCTION

Industrial radiography is a non-destructive method of looking for defects in materials, by examining the structures of welds, castings, and building components. Two types of radiation are used in industrial radiography: X-rays and gamma sources (such as iridium-192, cobalt-60, and selenium-75). Industrial radiography has a higher risk due to handling high activity sources [1]. In Korea, the average annual doses of industrial radiography workers are higher than in other groups of radiation practice fields. [2]. The Radiation of dose increase in the number of workers in Iran [3]. An excessively effective dose can cause health effects on radiation workers such as deterministic effects or stochastic effects. The radiation accident in 1999 involving two NDT operators in Taiwan caused diffuse hyperplasia of the thyroid gland, decreased blood lymphocyte counts, platelets, white blood cells, and also sperm cells [4].

Many Latin American countries such as Peru, Bolivia, and Peru report some radiological accidents on industrial radiography involving lots of radiation workers, the general public, and deaths in the world [5]. One of the causes of the accident is malfunction or damage to the gamma camera and supporting equipment [6]. Industrial radiography work is often carried out under difficult working conditions, such as inside confined spaces, in extreme cold or hot temperature, or during the night. Working under such adverse conditions might result in operational situations in which occupational radiation protection procedures may be compromised [7].

The annual dose limit for radiation workers has been set by The Regulatory Authority only 20 mSv per year or 100 mSv for a cumulative five years [8]. The operating organization or the company that carrying out the industrial radiography work must comply with the regulations and standards.

This paper showed the example of the accumulation of radiation doses from several industrial radiographers who may receive higher occupational doses and exceed the dose limit value. Afterward, it will be elaborated the safety actions that should be taken by the industrial radiography operating organization and the Regulatory Authority for workers to prevent the occurrence of deterministic effects in individuals and to ensure that all reasonable steps are taken to reduce the occurrence of stochastic effects of the workers at present and in the future.

MATERIAL AND METHODS

This paper uses test reports of individual dose monitoring collected over five years from 2011 to 2015 from four companies involved in industrial radiography practices in Indonesia. The companies were selected due to large numbers of workers and the radiation doses relatively higher than other companies in the same industry. Individual dose monitoring of workers is carried out trimonthly using thermoluminescence dosimeter (TLD) of CaSO4: Dy Indian BARC technology with the type of 1010 TLD reader by Nucleonix. The individual dose monitoring testing performed by Dosimetry Testing Laboratory that has been authorized by the Regulatory Authority and accredited to ISO-17025

RESULTS AND DISCUSSION

The dose data of radiation workers have been collected from four companies from 2011 to 2015. Company A has 212 radiation workers, Company B with 83 workers, Company C with a total of 257 people, and Company D has 336 workers. The results are obtained from the number of radiation workers for each year and the radiation worker code number is unique because it is only used for one person that cannot be given to another person.

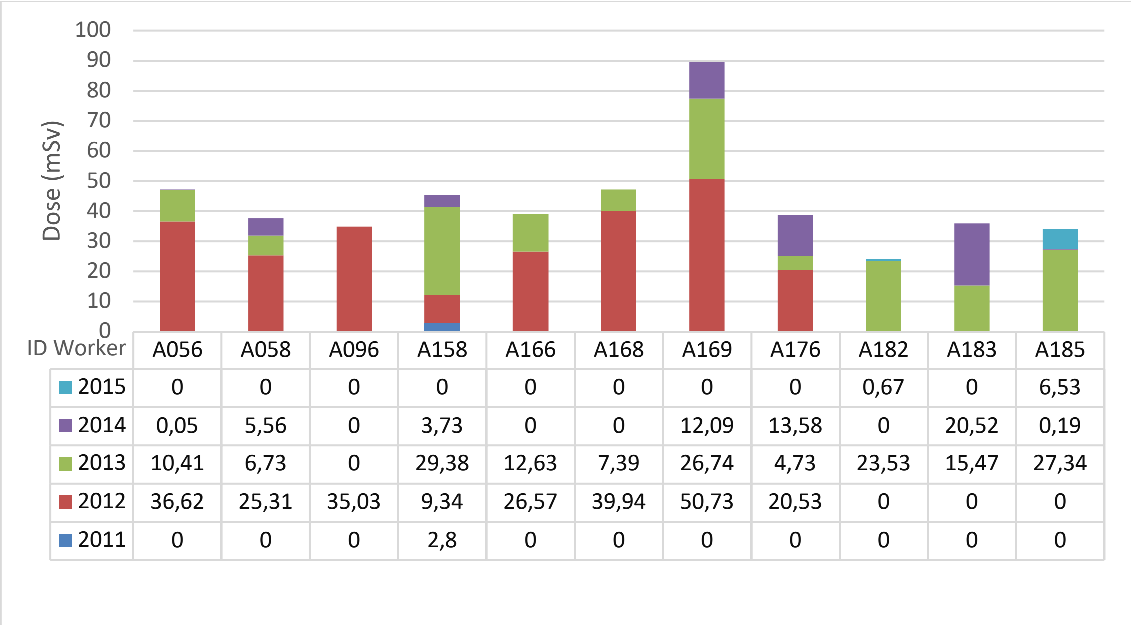


FIGURE 1. Accumulated doses of Company A in 2011 – 2015

FIGURE 1 shows that 11 people from all radiation workers of Company A had received a dose exceed the dose limit value and mostly in 2011. One of the seven radiation workers was received a dose of up to 50 mSv for a single year and in the following year the worker was still getting a dose that exceeded the dose limit value

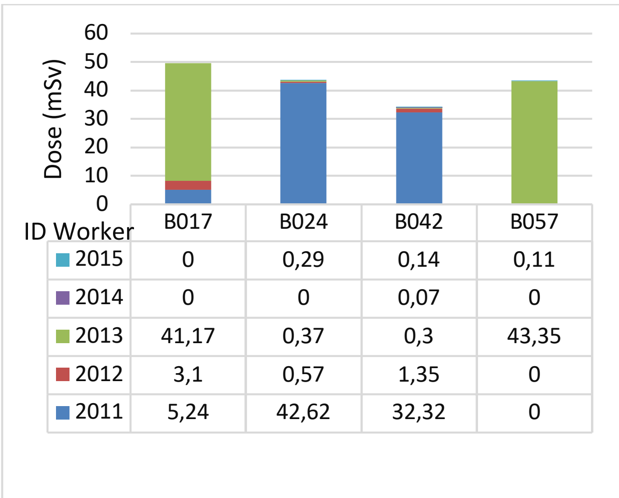
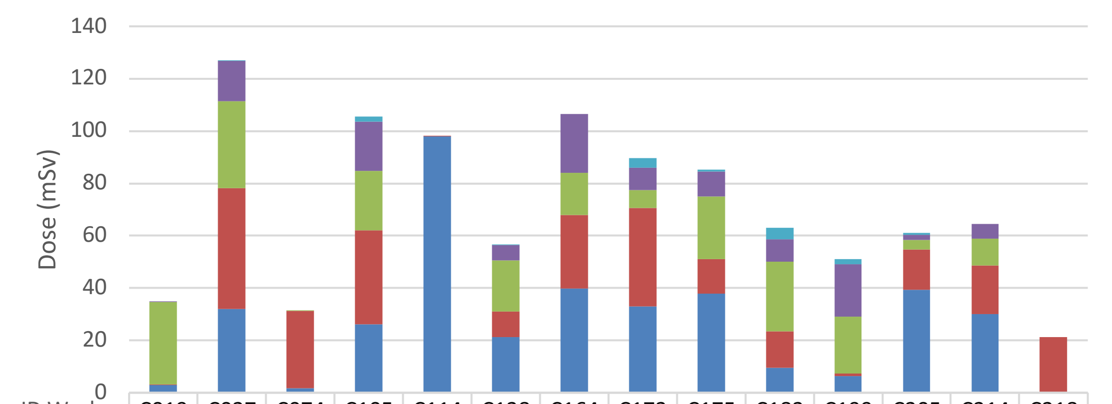


FIGURE 2. Accumulated doses of Company B in 2011 – 2015

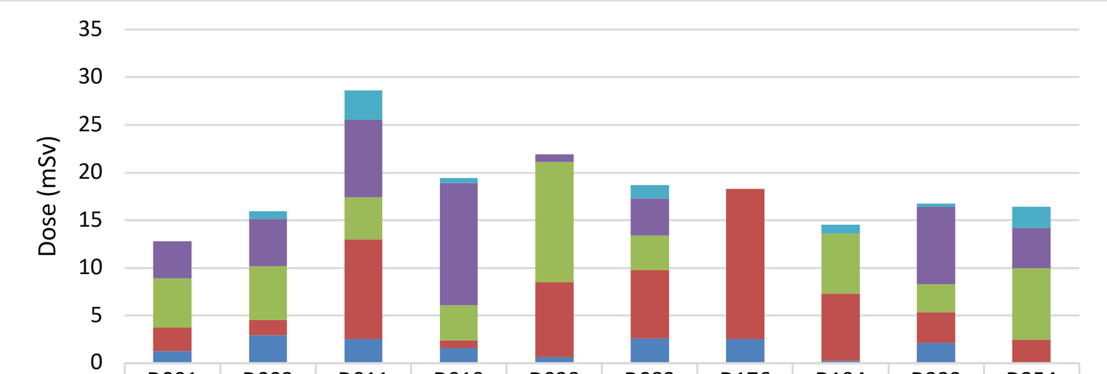
FIGURE 2 shows the radiation dose of workers for Company B. Two workers have exceeded the dose limit value but not for the following year.



ID Worker	C019	C027	C074	C105	C114	C138	C164	C173	C175	C183	C199	C205	C214	C218
2015	0	0,05	0	1,97	0	0,11	0	3,64	0,75	4,29	1,8	0,53	0	0
2014	0,06	15,39	0	18,73	0	5,82	22,51	8,58	9,61	8,62	20,16	2,17	5,39	0
2013	31,43	33,09	0,14	22,69	0	19,63	16,12	6,81	24,05	26,53	21,64	3,66	10,33	0
2012	0,31	46,16	29,66	36,12	0,06	9,69	28,19	37,55	13,08	14,14	1,04	15,19	18,49	21,31
2011	2,92	32,1	1,65	26,03	97,94	21,24	39,75	33,05	37,87	9,43	6,38	39,42	30,17	0

FIGURE 3. Accumulated doses of Company C in 2011 – 2015

FIGURE 3 shows exceeded doses at Company C. 14 workers had received an effective annual dose exceeding the dose limit value. There are some workers who three times in a row get a dose exceeding the dose limit value and some even get an effective dose for one year almost reaching twice a single dose limit value a year.



ID Worker	D001	D002	D011	D019	D028	D082	D176	D194	D223	D254
2015	0	0,81	3,11	0,55	0	1,41	0	0,99	0,36	2,21
2014	3,84	4,98	8,12	12,8	0,85	3,9	0	0	8,13	4,22
2013	5,2	5,66	4,4	3,72	12,57	3,59	0	6,29	2,91	7,51
2012	2,5	1,55	10,45	0,8	7,9	7,21	15,78	7,05	3,24	2,48
2011	1,23	2,96	2,56	1,59	0,62	2,59	2,51	0,23	2,13	0

FIGURE 4. Accumulated doses of Company D in 2011 – 2015

FIGURE 4 shows the accumulation of effective doses for five years from 10 radiation workers at Company D. There are no workers who have a dose that exceeds the dose limit value.

The accumulated effective dose from the four companies generally shows that the application of radiation safety regulations is quite maximal, but it cannot be stated that the regulation has been implemented. This needs further investigation. Obtaining high doses of workers caused by inadequate or damaged equipment, lack of application of radiation protection procedures, lack of education and training of workers, and overload of work [3] [9] [10].

The company should have implemented all applicable radiation safety regulations, supply and maintenance of industrial radiographic equipment and radiation protection equipment as well as in terms of occupational safety and health management such as employee training, workload systems, and others. The handling of workers who receive excess doses must be carried out such as periodic and special health checks. The company must also investigate the cause of workers receiving excessive doses of radiation.

Based on the Nuclear Safety Report issued by Indonesian Nuclear Energy Regulatory Agency (Bapeten) in 2015, from 91 institutions that use industrial radiography have been inspected by Bapeten, about 80% of institutions have fulfilled the safety and security provisions. However, the results of Bapeten's inspection showed that only 67% of industrial radiography operating organization had fulfilled the requirements regarding individual dose monitoring. That is because most institutions do not routinely record the results of individual dose evaluations on individual dose cards [11].

A regulatory system is needed to authorize an application involving sources of radiation to conduct radiography. The consequences of poor regulatory control can be serious and may result in hazardous conditions that may remain undetected for long periods. The general functions of the Regulatory Authority include the following: the development of radiography regulations and guidance; the assessment of applications for permission to conduct radiography; the authorization of such practices and the use of radiation sources associated with them, subject to certain specified conditions; the conduct of periodic inspections to verify compliance with the conditions; and the enforcement of any necessary actions to ensure compliance with the regulations and standards[7].

Compliance with safety regulations must also be accompanied by decisive action from the Bapeten at the time of inspection and following up on the overdose report submitted by the Dosimetry Testing Laboratory. Radiation safety on industrial radiography can be realized if all parties can work together.

CONCLUSION

The results of the accumulated effective dose of radiation workers on industrial radiography can be concluded that the industrial radiography operating organization still not compliant with the regulation and radiation safety aspects have not been applied maximally. The highest effective dose of radiation worker was 126.79 mSv which is accumulating for three consecutive years. The safety actions must be conducted by all parties to prevent the adverse health effect related to radiation exposure of the industrial radiography workers.

ACKNOWLEDGMENT

We would like to thank The Subdivision of Occupational Safety and Protection Radiation, Center for Technology Radiation Safety and Metrology (PTKMR BATAN) for providing the radiation dose data.

REFERENCES

[1] Canadian Nuclear Safety Commission, *Working Safely with Industrial Radiography*, PWGSC catalog number CC172-105/2014E-PDF ISBN 978-1-100-23558-5 (2014).
[2] J. Kim, S. Seo, D.N. Lee, S. Park, K.J. Im, *Occupational Exposure Characteristics and Factors Associated With Radiation Doses Among Korean Radiation Workers*, Radiation Protection Dosimetry (2020), Vol. 00, No. 00, pp. 1–8
[3] S. M. Hosseini Pooya, and M. R. Dashtipour, *Dosimetric Aspects of Optimization Of Protection In Iran Industrial Radiography Practice*, Radiation Protection Dosimetry (2018), pp. 1–6
[4] C. Ting, H. Wang, J. Lin, and C. Lin, *Evaluating The Radiation From Accidental Exposure During A Nondestructive Testing Event*," (2015).
[5] C. M. A. Lima and D. S. F. C. A, O*verview of Brazilian industrial radiography accidents with cutaneous radiation syndrome*, (2018) pp. 1–12,.
[6] International Atomic Energy Agency, *Lesson Learned from Accidents in Industrial Radiograph*, Safety Series No. 7, Vienna, (2003).
[7] International Atomic Energy Agency, *Radiation Protection and Safety in Industrial Radiography*, Safety Reports Series No. 13, (Vienna, 1999).

[8] Bapeten, Peraturan Kepala Bapeten no 4 tahun 2013 *Proteksi Dan Keselamatan Radiasi Dalam Pemanfaatan Tenaga Nuklir*. Jakarta, 2013.
[9] F. Mianji, S. M. H. Pooya, F. Zakeri, and M. R. Dashtipour, *A root cause analysis of the high occupational doses of industrial radiographers in Iran*, J. Radiol. Prot., vol. 184, p. 184.
[10] Jumpeno, B.Y.E.B. Evaluasi Hasil Pengujian Keselamatan Kamera Radiografi Gamma Industri Jenis Portabel di Laboratorium PTKMR BATAN Tahun 2012 - 2013. Widyanuklida, Vol. 14 No. 1, November 2014: 13 - 21
[11] Bapeten, Laporan Keselamatan Nuklir 2015. Jakarta, 2016.

National Approaches of Cybersecurity for Nuclear Facilities in Indonesia

Mulyadi[1, a)] and Dwi Rahayu[2, b)]

[1]*Directorat of Government Protection, National Cyber and Crypto Agency*
[2]*Neutron Scattering Laboratory, National Nuclear Energy Agency*

[a)] mulyadi@bssn.go.id
[b)] dwi27@batan.go.id

Abstract. In the digital era as it is today, the application of internet-based technology (cyber) in nuclear facilities cannot be avoided even though the security risks are very vulnerable to exploitation. Based on the 2018 NTI report, Indonesia has a nuclear cybersecurity score below the average of other countries in the world, even though this score has not increased since 2016. This paper aims to analyze the current status of cybersecurity for nuclear facilities in Indonesia and its application from a National Government perspective using SWOT analysis. Based on the study, it is known that the implementation of cybersecurity in nuclear facilities in Indonesia has not been done, this can be seen from the absence of regulations covering cybersecurity in the nuclear field. Nuclear security is assessed from the side of nuclear stakeholders only as far as physical security, while cybersecurity is judged from the perspective of cybersecurity has not stated specifically on nuclear facilities. Based on the results of the SWOT analysis that was conducted, Indonesia has several weaknesses but also has strengths that can be empowered to resolve existing weaknesses and threats. Several concrete steps need to be taken that are written in the recommendations of this study.

Keywords: cybersecurity, nuclear, SWOT

INTRODUCTION

In the digital era as it is today, the application of internet-based technology (cyber) in nuclear facilities cannot be avoided even though the security risks are very vulnerable to exploitation. Nuclear facilities are one of the national vital objects that require special attention in terms of security and safety, including the application of internet-based technology (cyber) that installed in facilities.

The cyber incident occurred in the United States in 2003, was infected by a worm type virus called a slammer and killed the security perimeter at a nuclear facility. Then in 2010, in Iran's nuclear facilities, a Stuxnet virus attack occurred that infects via a USB drive from one of the devices and then spreads to the network (Dine et al., 2016).

Although until now there have been no reports of cyber attacks on Indonesian nuclear facilities, according to (Iwan Sumantri, 2016), in 2015 Indonesia experienced a total of 28,430,843 cyber attacks with details on the website hacking incidents 13,955, malware activity 461,511, information leak. vulnerability 28,657, data leak and manipulation 8,134, and the most targeted domain is .go.id. In 2016 the total number of attacks increased significantly, namely 135,672,984 cyber attacks, consisting of 47% malware, 44% fraud, 4% vulnerability, and 1% intrusion.

FIGURE 1 shows the trend of cyber attacks from 1990 to 2016 has increased and will continue to increase as more and more implementations of internet-based technology on nuclear facilities. For this reason, the aspect of cybersecurity in nuclear facilities is a very important thing to be applied.

In Indonesia, Nuclear Research is developing very rapidly. Based on the law, the National Nuclear Energy Agency of Indonesia (BATAN) is an R&D agency and the Nuclear Energy Regulatory Agency (BAPETEN) is a regulator. Of course, the application of cybersecurity to its nuclear facilities should not be managed carelessly.

Currently, the implementation of nuclear cybersecurity in Indonesia is not good enough to meet global challenges, this is indicated by the absence of basic regulations such as strategies, standards, policies which focus on regulating nuclear cybersecurity. In addition, based on the 2018 NTI report, Indonesia has a nuclear cybersecurity score below the average of other countries in the world, even this score has not changed since 2016.

This paper analyzed the current status of cybersecurity for nuclear facilities in Indonesia and its application from a National Government perspective using SWOT.

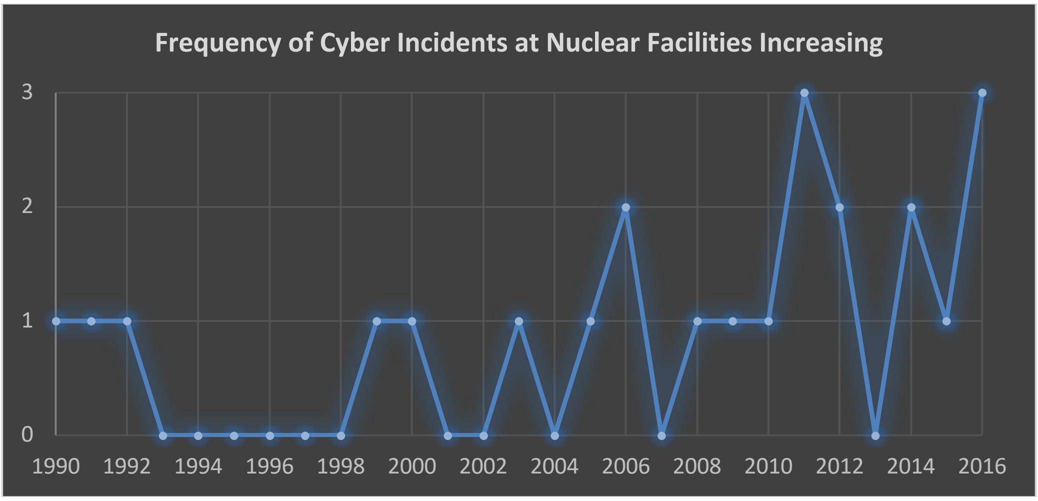


FIGURE 1. The number of cyber incidents at nuclear facilities 1990 - 2016
Source: (Dine et al., 2016)

In the last chapter, we proposed the recommendations and discussed the steps to accelerate the adoption of good cybersecurity in nuclear facilities in Indonesia and also discussed the strengthening of the implementation of nuclear cybersecurity in Indonesia from the perspective of the National Government. So that the damage caused by the weak cybersecurity system at nuclear facilities can be avoided as early as possible.

URGENCY OF CYBERSECURITY IN NUCLEAR FACILITIES

In general, we are familiar with the two terms of information security and cybersecurity. Many definitions of these two things are seen from their perspective and scope. Information Security makes information as a secure object, because of that discussion of information security includes the management of information security both digital information and physical information based on a "paper and pencil" basis. But on the contrary, Cyber Security only addresses security within the cyber or virtual sphere.

In this discussion, the concept used is cybersecurity because the object of security is a nuclear facility that is internet-based (cyber) or virtual. The incidents that occur at nuclear facilities are not limited to theft of information or spy, but further in the form of sabotage that affects the function of a number of nuclear facilities instruments experiencing malfunction or damage, causing material losses and even fatalities such as the following cases (Dine et al., 2016):

- Davis-Besse Nuclear Power Station, US, 2003
  Around 75,000 servers around the world are infected by a worm called a slammer in 10 minutes. This virus attack has also occurred at Davis-Besse Nuclear Power Station. This is caused by the process control system on the plant connected to an unsecured internet network so that it can easily be exploited, then spread malicious programs. The slammer worm uses a lot of bandwidth capacity and causes the plant's safety parameter display system (SPDS) down for 4 hours.
- Natanz Fuel-Enrichment Plant, Iran, 2010
  Stuxnet virus attack on Iran's nuclear facilities some time ago became a worldwide concern, this virus spread into the control instruments of Iran's nuclear facilities via USB drive. The first occurred in 2005, the virus attacked Siemens programmable logic controllers (PLCs) at Iran's Natanz uranium enrichment facility and tried to screw up the uranium enrichment by closing valves that inserted uranium hexafluoride gas into centrifuges. Then the second occurred in 2009 and exploited in 2010 trying to disrupt the process by changing the rotational speed of centrifugal gas in Natanz.
- Oak Ridge National Lab, US, 2011
  The incident in ORNL, the attacker carried out the attack by finding vulnerabilities contained in the internet explorer application, the attacker spread phishing e-mails to users on behalf of the human resort department. An email containing an address / URL that connects to a malicious website. When a user accesses the URL, it automatically downloads malware that was deliberately set up by the attacker.
- Korea Hydro and Nuclear Power Company, South Korea, 2014
  Hackers in the name of anti-nuclear hacked the company's network and stole blueprints and manuals for two nuclear power plants through phishing e-mail. It is estimated that hackers sent 5,986 phishing emails

containing malicious codes to 3,571 employees of nuclear plant operators. Hackers have claimed that they have 10,000 employees' personal information, as well as a nuclear facility electrical installation scheme.

- University of Toyama Hydrogen Isotope Research Center, Japan 2016
  In June 2016 it was discovered that hackers used a spear-phishing attack to steal personal data and research data from the University of Toyama's Hydrogen Isotope Research Center. Hackers have provoked university students/researchers by asking several questions and sending documents that have been injected with malware to researchers via email to commit data theft.
- Gundremmingen Nuclear Power Plant, Germany, 2016
  In April 2016, it was reported that computers at the Gundremmingen Nuclear Power Plant were infected with W32. Ramnit and Conficker type malware that could potentially cause data theft and remote control. This malware is found on devices installed by the application to simulate the movement of nuclear fuel rods.

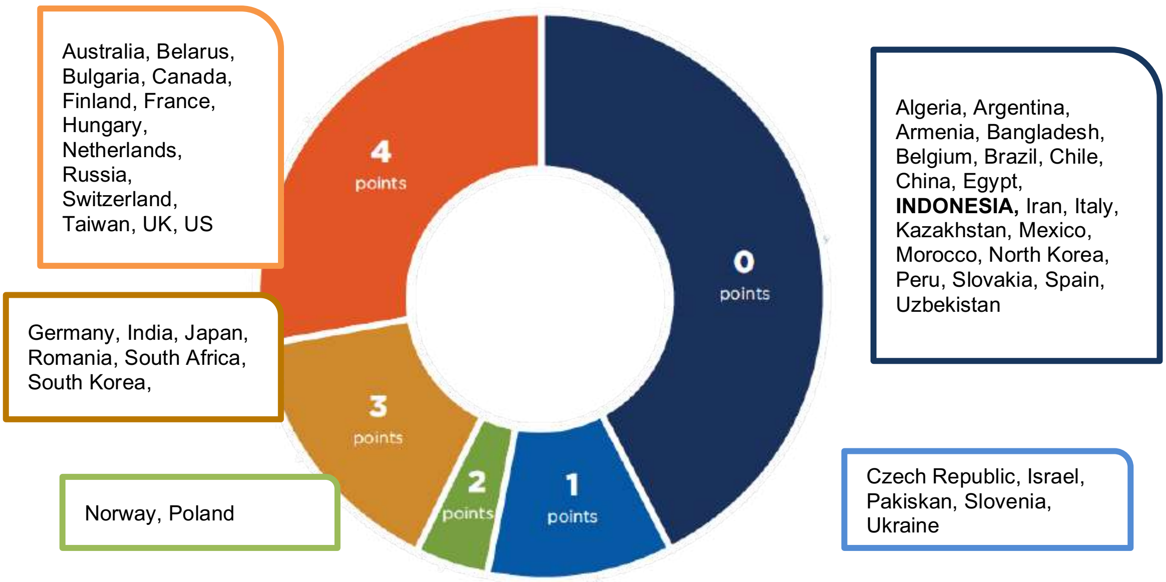


FIGURE 2. NTI Nuclear Security Index Cyber Scores

FIGURE 2 above shows that Indonesia is among the countries that have not implemented the security perimeter measured by four (4) questions such as:

- Does the country require nuclear facilities to be protected from cyberattack?
- Does the country require nuclear facilities to identify critical digital assets?
- Does the country incorporate cyber threats into its design basis threat or other threat assessment?
- Does the country require performance-based testing of its cybersecurity measures?

TABLE 1. Nuclear Security Indonesia's Profile by Category

Category		2018 Score	Change Since 2016
Number of Site		80	0
Security and Control Measure		64	0
	On-Site Physical Protection	100	0
	Control and Accounting Procedures	100	0
	Insider Threats Prevention	33	0
	Response Capabilities	71	0
	Cybersecurity	20	0
Global Norms		81	+5
	International Legal Commitments	100	0
	Voluntary Commitments	60	+20

	International Assurances	75	0
Domestic Commitments and Capacity		100	+16
	UN Security Council Resolution (UNSCR) 1540 Implementation	100	+60
	Domectis Nuclear Security Legislation	100	0
	Independent Regulatory Agency	100	0
Risk Environment		44	+4
	Political Stability	60	-5
	Effective Governance	38	0
	Pervasiveness of Corruption	25	+25
	Group(s) Interested in in Committing Act of Nuclear Terrorism	50	0

(red)	Below average
(orange)	Average
(green)	Above average

Based on the 2018 NTI survey, Indonesia is ranked 27th (in sabotage ranking) for the category of countries that have non-weapons nuclear facilities. In the details above, it can be seen that the weakest (red block) aspects of Indonesia are the Security and Control Measure, consisting of the insider threat prevention and cybersecurity points, as well as the Risk Environment aspects including Political Stability, Effective Governance, Pervasiveness of Corruption, and Groups Interested in Committing Act of Nuclear Terrorism.

TABLE 1 also shows that nuclear facilities in Indonesia still have security and safety risks that have not been handled properly. These points refer to one source of problems, namely the lack of management of the nuclear field by the state which causes weak aspects of the Risk Environment, especially in the field of cybersecurity and insider threat prevention (indicating by red colour).

CYBERSECURITY AT NUCLEAR FACILITIES IN INDONESIA

In this paper, the authors use an approach instrument from Cyber Security at Nuclear Facilities: National Approaches An ISS Research Project in Cooperation with the Nuclear Threats Initiative (NTI) that has been published by *Fachhochschule Bradenburg University of Applied Sciences Institute for Security and Safety* (Holl et al., 2015).

1. National Legislation
   National legislation as the highest level of reference for the implementation of cybersecurity in nuclear facilities specifically, which means that it is separate from national nuclear legislation in general such as nuclear power, radioactive material, and others. Based on studies the aspect of national legislation are good. There is institution carrying out business in the nuclear and security sectors.
   In the field of nuclear facilities, there is the Nuclear Energy Supervisory Agency (BAPETEN) as the regulator, and the Nuclear Energy Agency (BATAN) as the implementation of research and development of nuclear science and technology in Indonesia. Meanwhile in the security sector, the government established the National Cyber and Crypto Agency (BSSN) as the organizer of security in Indonesia.
2. Regulatory Framework
   In the case of the Regulatory Framework, it explains the application of cybersecurity in nuclear facilities precisely and clearly. The entities were responsible for forming dynamic frameworks so that they can meet the requirement of the development of the field of nuclear cybersecurity.
   Based on the studies that have been carried out, the results show that coordination or synergy between government institutions, in this case BAPETEN, BATAN, and BSSN in the implementation of cyber security at nuclear facilities, n terms of planning, implementation, and supervision are still lacking.
   The government has already established the necessary institutions, but these institutions are still moving partially or separately, thereby slowing down the process of implementing cybersecurity at nuclear facilities.
3. Regulation and Guidance
   This chapter discusses regulations, standards, guidelines for implementing legislation that is concise, clear, and detailed to become rules for implementing cybersecurity in nuclear facilities. based on instruments from Cyber Security at Nuclear Facilities: National Approaches An ISS Research Project in Cooperation with the Nuclear Threats Initiative (NTI) shows that the government's weakness is in the ragulation sector and the guidelines used as the basis for implementing cyber security at nuclear facilities.

The regulation is very important because it is a cybersecurity guideline for nuclear facilities, if these documents do not exist, planning, implementation or monitoring of cybersecurity at nuclear facilities in Indonesia cannot be carried out because there are no binding factors, determination of duties and responsibilities, rights, and the obligations of all parties in the implementation of cyber security in Indonesia.

4. Licensing
   Nuclear cybersecurity is implemented based on designs that are made and certainly run according to the design continuously. All parameters stated on the instrument are not fulfilled because cybersecurity at nuclear facilities does not appear to be a major concern. This causes cybersecurity criteria at all stages of design for the construction of nuclear facilities to be included. On the other hand, the regulations for discussing cybersecurity at nuclear facilities have not been well worked out. Therefore, this aspect is also the weakest aspect in Indonesia.
5. Associate Regulatory Activities
   Supporting matters related to nuclear cybersecurity such as supply chain control to personal security to law enforcement training, and other matters that have an impact on nuclear cybersecurity. In Indonesia, regulations relating to cybersecurity at nuclear facilities are still weak. This is indicated by the absence of an assessment and supervision of cyber security at nuclear facilities.
6. Education
   This chapter contains education that specifically discusses nuclear cybersecurity both through formal educational institutions such as universities, as well as courses, training, seminars, socialization, and others.

In addition to trainings that are carried out independently by institutions for employees, there is also training or information sharing conducted by the government to the general public. Not only that, the government has schools that focus on developing nuclear science and technology, namely the Nuclear Technology College (STTN), and a cyber security development school, namely the National Cyber and Crypto Polytechnic (PSSN).

SWOT ANALYSIS (STRENGTH – WEAKNESS – OPPORTUNITY – THREATS)

SWOT analysis is carried out to get the profile of nuclear cybersecurity in Indonesia nowadays. The SWOT analysis was carried out with the 6 instrument items above. Based on the data collected, the results of the SWOT analysis are as follows:

TABLE 2. SWOT Analysis of the Condition in Indonesia

Internal Factor / **Eksternal Factor**		**STRENGTH**	**Weakness**
		✓ **Legal/Government Institution** ✓ **Availability of Budget** ✓ **Power**	✓ **Low Human Resource Competency** ✓ **No Regulation, Standards, Policy of cybersecurity** ✓ **Bureaucracy**
✓ **Opportunities**	✓ **International or Regional Organization Support**	✓ The authorized institution in cybersecurity consists of national organizations, namely government institutions, in this case, the National Cyber and Crypto Agency in charge of cybersecurity in Indonesia, and the BATAN and BAPETEN in charge of developing nuclear regulations and science and technology.	✓ Reducing complexity, the more complex the bureaucratic process, procedures, operations, the more there is the possibility of vulnerability loopholes. ✓ Collaboration IAEA – Education/ Training/ Benchmarking
✓ **Threats**	✓ **Insider Threats** ✓ **Outsider Threats – Nuclear Terrorism, Hacking, Sabotage** ✓ **Corruption Sectoral ego**	✓ Implementing Active Defense techniques. Active Defense does not have the active intention to carry out attacks, but is active in anticipating new attacks by always updating the latest cybersecurity	✓ Transformation means changing completely, starting from the quality and capacity of human resources, business processes, work programs, and so on.

The results of the SWOT analysis show that through a national approach, Indonesia has the characteristics shown in **TABLE 2**.

1. Strength
   Indonesia has a legal institution that manages nuclear technology and cybersecurity nationally. In the nuclear sector, Indonesia has BATAN that conducts research and development of nuclear technology for agriculture, health, industry, energy, etc. In addition, Indonesia also has BAPETEN, which deals with national nuclear regulations. Then, in 2017, Indonesia established the National Siber and Crypto Agency (BSSN) which specifically manages national cybersecurity. The three government agencies must work together to support each other in building cybersecurity in nuclear facilities.
   As a government agency, BATAN-BAPETEN-BSSN should have a sufficient budget to support programs of implementation of nuclear cybersecurity. Also, it has the power as a government institution to make several rules and policies to foster and regulate stakeholders with an interest in the nuclear field.
2. Weakness
   The issue of cybersecurity has recently become the concern of various parties, both governments, private, and academic, along with the development of information technology. This situation is not balanced by the formation of human resources who are ready to face the use of ICT properly, correctly, and safely in nuclear facilities. For this reason, it is necessary to accelerate the improvement of the quality of nuclear human resources so that they are more concerned about working safely in the cyber area.
   Government institutions are also still constrained by the complexity of the bureaucracy that needs to be passed to implement certain programs or policies, thus slowing down the process of implementing cybersecurity in nuclear facilities. The length of this bureaucracy also has an impact on the slow production process of regulations, policies, and guidelines on nuclear cybersecurity in Indonesia. For this reason, bureaucratic pruning is needed so that acceleration occurs in the process of coordination, sharing, joining programs, and so on in developing nuclear cybersecurity.
3. Opportunity
   One thing that is always owned by government agencies is easy to get access to anywhere, including international organizations such as the IAEA (International Atomic Energy Agency). BATAN / BAPETEN as a symbol of the Indonesian government must be able to take advantage of opportunities to collaborate in various fields of nuclear development including in nuclear cybersecurity including training, courses, benchmarking, national strategies, and others.
4. Threats
   Being a threat that needs to be seriously managed is insider threats. Brave breakthroughs and appropriate strategies are needed to find insider threats to determine how to solve them. Finding insider threats is more difficult than finding outsider threats because insider threats are on the internal side, so it is difficult to detect them. Therefore, an insider threat prevention policy is needed in an organization.

Another threat that is also very detrimental and impedes cybersecurity is the level of corruption in Indonesia. Corruption itself is one form of insider threat that is difficult to resolve. Likewise, with sectoral egos, this greatly impedes the process of cooperation that should be able to accelerate the development of nuclear cybersecurity in Indonesia.

RECOMMENDATION

The recommendation as the result SWOT Analysis Generally divided into 4 points (Dine et al., 2016), it can be said:

1. The authorized institution in cybersecurity consists of national organizations, namely government institutions, in this case, the National Cyber and Crypto Agency in charge of cybersecurity in Indonesia, and the BATAN and BAPETEN in charge of developing nuclear regulations and science and technology. These three Institutions are obliged to work together in particular in 2 (two) ways, namely compiling, implementing, and supervising cybersecurity regulations on nuclear facilities as well as forming and shaping human resources that have the capacity in cyber nuclear security.
   In addition to national cooperation, it must also be actively involved in international events with the IAEA, especially in the development of human resources, and the process of establishing regulations on cybersecurity in nuclear facilities such as dialogue, training, forums, and so on.
2. Implementing Active Defense techniques. Active Defense does not have the active intention to carry out attacks but is active in anticipating new attacks by always updating the latest cybersecurity techniques,

vulnerability information through discussion forums with experts, communities, etc. others both between government sectors in Indonesia and with international organizations.

3. Reducing complexity, the more complex the bureaucratic process, procedures, operations, the more there is the possibility of vulnerability loopholes. Therefore, reducing the complexity of all matters relating to nuclear facilities is very important, so that it is easier to conduct supervision and control. In government institutions, bureaucracy is generally very long starting from the administration, execution of implementation, and accountability reports of activities that hinder the work process, especially the process of cooperation, coordination that demands to be quickly carried out by an institution, or between work units within one institution.
4. Transformation means changing completely, starting from the quality and capacity of human resources, technology, budgets, business processes, work programs, and so on. All of these things were overhauled and replaced with new ones. Human resources are upgraded, technology is upgraded to the latest version, work programs are made as innovative as possible and not monotonous from year to year, budget efficiency, management improvements, and business processes so that cybersecurity at nuclear facilities is well implemented. This transformation also aims to overcome the threats and weaknesses faced by Indonesia in accordance with the results of the SWOT analysis conducted. Transformation removes sectoral egos, closes gaps for insider and outsider threats, and of course eradicates corruption.

in more detail, the recommendations are shown in **TABLE 3.**

Table 3. Recommendations for Application of Cyber Security to Nuclear Facilities in Indonesia

	DOMAIN			
	Institutionalize Cybersecurity	**Mount an Active Defense**	**Reduce Complexity**	**Pursue Transformation**
Governments and Regulators	• Prioritize development and implementation of regulatory framework • Draw talented people into the cyber nuclear field	• Enhance cyber expertise within governmental and regulatory bodies • Consider how to develop and exercise cyber incident response capabilities • Support efforts to re-tool defense strategies and promote information sharing between governments	• Provide financial, personnel, and research support to efforts to minimize complexity in critical facility systems	• Invest in augmenting human capacity, research, and development in the cyber-nuclear space
International Organizations	• Support, through international dialogue and definition of relevant best practice, international cooperation and an expanded focus on cybersecurity on nuclear facilities • Develop and provide guidance and training to governments and facilities as requested	• Facilities sharing of threat information, where possible and as appropriate	• Provide platforms for discussing and developing solutions for reducing complexity	• Foster innovation and continue to think creatively about how to mitigate this threat • Enlist a variety of voices and perspective to join the conversation

CONCLUSION

Based on research that has been done it is known that the implementation of cybersecurity in nuclear facilities in Indonesia has not been done, this can be seen from the absence of regulations covering cybersecurity in the nuclear field. Nuclear security is assessed from the side of nuclear stakeholders only as far as physical security, while cybersecurity is judged from the perspective of cybersecurity has not touched specifically on nuclear facilities. A number of concrete steps need to be taken, based on the results of the SWOT analysis that conducted, Indonesia has several weaknesses, but also has strengths that can be empowered to resolve existing weaknesses and threats. Several recommendations are proposed to accelerate the implementation of nuclear cybersecurity in Indonesia, which include authorized institutions in cybersecurity, Implementing Active Defense techniques, Reducing the complexity of bureaucracy, and upgrade system of quality and capacity of human resources, technology, budgets, business processes, work programs, and so on.

REFERENCES

[1] Law Number 10 of 1997 on Nuclear Power.

[2] Government Regulation Number 54 of 2012 on Safety dan Security Nuclear Installation.

[3] Presidential Regulation Number 53 of 2017 on Establishmnet of the National Cyber and Crypto Agency revised by Presidential Regulation Number 133of 2017.

[4] Presidential Regulation Number 71 of 2001 on Establishment of National Nuclear Technology Institute.

[5] Head of National Cyber and Cryopto Agency Regulation Number 12 of 2019 on Organizational Structure of National Cyber and Crypto Polytechnic.

[6] Head of National Cyber and Cryopto Agency Regulation Number 2 of 2018 on Organizational Structure of National Cyber and Crypto Agency.

[7] Head of National Nuclear Energy Agency Number 14 of 2013 on Organizational Structure of National Nuclear Energy Agency

[8] Cyber Security at Nuclear Facilities: National Approaches An ISS Research Project in Cooperation with the Nuclear Threats Initiative (NTI), First Edition (Fachhochschule Bradenburg University of Applied Sciences Institute for Security and Safety, Juni 2015).

[9] Building a Framework for Assurance, Accountability, and Action, Fourth Edition (NTI Nuclear Secutity Index Theft Sabotage, September 2018).

[10] Alexander Van Dine, Michael Assante and Page Stoutland, Outpacing Cyber Threats Priorities for Cybersecurity at Nuclear Facilities, (Nuclear Threat Initiative, 2016).

[11] Ju-min Park and Meeyoung Cho, 17 March 2015, https://www.reuters.com/article/us-nuclear-southkorea-northkorea/south-korea-blames-north-korea-for-december-hack-on-nuclear-operator-idUSKBN0MD0GR20150317.

[12] Sumantri Iwan, Tren Serangan Siber Nasional 2016 dan Prediksi 2017. in National Cyber Security Defense, 2017 (pp 1-35), Kemenkominfo.

[13] Gurel Emet, SWOT Analysis: A Theoritical Review, The Jurnal of International Social Research Volume 10, (Augutust 2017).

[14] Freddy Rangkuti, Analisis SWOT Teknik Membedah Kasus Bisnis, (Jakarta, PT. Gramedia Pustaka Utama, 2006).

Security Management System for Nuclear Utilization Sector

Fery Putrawan Cusmanri[1, a)]

[1]Badan Pengawas Tenaga Nuklir Indonesia

[a)] email: f.putrawan@bapeten.go.id

Abstract. The issue of security is one currently hot issue. According to Government Regulation Number 54 Year 2012, one of the obligations of the license holder is to establish a security management system. However, the Nuclear Energy Regulatory Agency (BAPETEN) regulations related to the security management system have yet to be published. This paper was made to provide a proposal for the preparation of a security management system document that can be used by license holders and BAPETEN as the regulatory body. In compiling a document management system document, the reference TDL-004 - Nuclear Security Management for Research Reactors and Related Facilities is completed with several other international standards that have already been established about management or security systems, namely ISO 27001: 2013 - Information Security Management System and GSR Part 2 - Leadership and Management for Safety. From the results of the preparation it was found that several important aspects to consider in the management system document are security objectives and how to achieve them, graded approach, leadership in security, security culture, resource management, operational security, security units, and performance assessment and potential for improvement.

Keywords: security, management system, leadership, regulation.

INTRODUCTION

In the year 2019, IAEA Incident and Trafficking Database tell a fact that there are 189 reports of nuclear incident in 36 countries around the world, signaling that illegal activities and events related to nuclear and other radioactive materials, including tradings and malicious act, still happen [1].

Concern over nuclear security issues as a very important issue marked by the held of the Convention on the Physical Protection of Nuclear Materials (CPPNM) which was signed in Vienna on March 3, 1980, before being amended in 2005.

National regulations for nuclear security in Indonesia acted on Government Regulation Number 54 of 2012 concerning Nuclear Installation Safety and Security for Nuclear Materials and Installations cluster with further arrangements regarding physical protection in the BAPETEN Chairman Regulation (BCR) No. 1 Year 2009 concerning Provisions for Protection Systems Physical Installation and Nuclear Materials and Government Regulation No. 33 Year 2007 concerning Safety of Ionizing Radiation and Security of Radioactive Sources which the security aspects are further regulated by BCR No. 6 Year 2015 concerning Security of Radioactive Sources for Radiation Facilities and Radioactive Materials cluster.

According to Article 62 paragraph (1) of Government Regulation No. 54 Year 2012 concerning Nuclear Installation Safety and Security, one of the obligations of a license holder is to establish and implement a nuclear installation security management system [2]. According to the IAEA security glossary, what is meant by nuclear security is [3] :

Prevention, detection, and response to criminal or intentional acts involving or directed at nuclear materials, other radioactive materials, related facilities, or related activities.

Prevention and detection and response to, theft, sabotage, unauthorized access, illegal displacement, or other malicious actions involving nuclear material, other radioactive material, or related facilities. It should be noted that 'nuclear security' includes 'physical protection' because the term is understood from a consideration of the Physical Protection Objectives and Basic Principles, CPPNM and Amendments to CPPNM.

According to Government Regulation No. 33 Year 2007 concerning Safety of Ionizing Radiation and Security of Radioactive Material, what is meant by the security of radioactive sources is "actions taken to prevent unauthorized access or destruction, and loss, theft, and/or illegal transfer of Radioactive Sources." [4].

According to Article 4 paragraph (2) of Act Number 10 Year 1997 concerning Nuclear Energy, one of the nuclear supervision tasks mandated to the regulatory body is to draft regulations[5]. The drafting of the regulations themselves is carried out by revising existing regulations to keep them in line with social and technological developments or making new regulations to fulfill the mandate of higher regulations.

In Article 63 of Government Regulation No. 54 Year 2012 concerning Nuclear Installation Safety and Security, there is a mandate related to the making of BAPETEN Regulation concerning the nuclear installation security management system. However, until now BAPETEN Regulation related to management systems is still limited to safety aspects and has not discussed security aspects.

Therefore, a BAPETEN Regulation related to the security management system is needed as a guideline for license holders to draw up a security management system document to meet the requirements of Government Regulation No. 54 Year 2012 concerning Nuclear Installation Safety and Security.

This paper is made to provide recommendations that can be used in drafting BAPETEN Regulation on nuclear security management systems. However, the recommendations presented in this paper are general and not limited to nuclear installations so that they can be implemented in any nuclear power utilization activity. The method used in writing this paper is the study of literature using international and national standards related to management systems that are enhanced by discussions with several parties.

THE PROBLEM BASIS

General Structure of Management System

The general structure of the recommended management system clause was adopted from Annex SL which is a high-class structure of the international ISO standard [6]. This approach is recommended that the security management system will facilitate parties who have already used other international standards in their management systems.

As general, Annex SL consists of 10 clauses:

TABLE 1. 10 (Ten) Clauses of Annex SL

Clause	Sub-clause
Clause 1 - Scope	
Clause 2 - Normative Reference	
Clause 3 - Terms and definitions	
Clause 4 - Context of the organization	4.1 - Understanding the organization and its context 4.2 - Understanding the needs and expectations of interested parties 4.3 - Determining the scope of the management system 4.4 - Management system
Clause 5 - Leadership	5.1 - Leadership and commitment 5.2 - Policy 5.3 - Organisation roles, responsibilities and authorities
Clause 6 - Planning	6.1 - Actions to address risks and opportunities 6.2 - Management system objectives and planning to achieve them
Clause 7 - Support	7.1 - Resources 7.2 - Competence 7.3 - Awareness 7.4 - Communication 7.5 - Documented information
Clause 8 - Operation	8.1 - Operational planning and control
Clause 9 - Performance Evaluation	9.1 - Analysis and evaluation 9.2 - Internal audit 9.3 - Management Review
Clause 10 - Improvement	10.1 - Non-conformity and corrective action 10.2 - Continual improvement

The clause about leadership is one of the clauses which is quite influential on the standards or other rules related to the management system for example General Safety Requirements (GSR) Part 2 - Leadership and Management for Safety which one of the requirements is that senior management also has responsibilities related to safety [7] which was later adopted in the draft of BAPETEN Regulation regarding Nuclear Installation Management Systems and the Use of Ionizing Radiation Sources.

According to Article 62 paragraph (2) of Government Regulation No. 54 Year 2012 concerning Nuclear Installation Safety and Security, the security management system contains at least [2]:

1. Security culture;

2. Ranking and documentation;
3. Management responsibility;
4. Resources management;
5. Process implementation; and
6. Measurement of effectivity, assessment, and improvement opportunity.

In preparing this recommendation, a reference related to the nuclear security management system is the IAEA publication related to security, namely TDL-004 - Nuclear Security Management for Research Reactors and Related Facilities. Although in general, this reference is talking about a research reactor, the mindset and ideas contained in it can be taken and adopted to be a picture of a nuclear security management system that can be applied.

Besides TDL-004 other references are also used, namely GSR Part 2 - Leadership and Management for Safety and ISO 27001: 2013 - Information Security Management System. Both of these references are used to enrich and perfect the recommendations made.

In the end, the recommendation for a nuclear security management system prepared is a brief description of TDL-004 with the addition of several clauses from GSR Part 2 - Leadership and Management for Safety, ISO 27001: 2013 - Information Security Management System and the contents stipulated in the Regulations Government Number 54 of 2012 concerning Nuclear Installation Safety and Security and was built by adopting the Annex SL structure.

Interface with the Safety Management System

Diversity of Safety System

According to Article 9 paragraph (2) letter e Government Regulation Number 54 Year 2012 concerning Nuclear Installation Safety and Security, one of the basic principles of nuclear plant safety is the diversity [2]. In the explanation of the same regulation, what is meant by diversity is an effort to diversify structures, systems, and components that have identical functions, but have different characteristics, to reduce the possibility of failure with the same cause. High diversity makes the complexity of the process even higher so that it will be difficult for the enemy to do sabotage or hacking the system.

Graded Approach

One of the requirements of GSR Part 2 - Leadership and Management for Safety is a graded approach, which includes considering the level of danger and potential impacts related to safety[7]. This hazard level and potential impact on safety analysis can help in ranking security priorities in a facility.

RESULTS AND DISCUSSION

From the results of the literature study, several clauses need to be considered for inclusion in the requirements of a nuclear security management system:

Security Objectives and How to Achieve It

The licensee as senior management must establish what the security objectives are and how to achieve these security objectives. This is important so that the organization can determine the needs and ways to achieve these goals and so that every party in the organization can find out.

Graded Approach

The licensee must take a graded approach to make decisions related to security. The licensees must rank security-related risks and adjust actions based on the rating that has been done. Risk ranking can be carried out based on potential threats that may be present in each specific activity or location at the facility.

For example, for radiation facilities and radioactive material clusters, according to the BCR No. 6 Year 2015 concerning Security of Radioactive Sources the security levels for exporting, importing, using, producing radioisotopes and managing radioactive waste are grouped into three security levels (A, B, and C) where the levels are categorized according to the activity/value D and the detail of the activities carried out by the security. From rating A to C, then the security measures are adjusted. As explained in article 40 of the same regulation, that for security level A, security measures using deferment equipment include at least 1 (one) electronic key and 2 (two) manual keys. Whereas at security level B, at least one (1) manual key is installed at the entrance of a fixed or storage facility.

Leadership on Security

The licensee as senior management has the highest responsibility related to security. The licensee must determine that nuclear security has a high priority and must demonstrate and maintain security culture within his organization.

Every manager at each level also has the responsibility to demonstrate and maintain security culture in his working environment.

These things are important because the leadership commitment and its application have a great influence on the quality of employee performance [8].

Security Culture

Based on NSS 7 - Nuclear Security Culture, nuclear security culture is a collection of characteristics, attitudes, and behavior of individuals, organizations, and institutions that function as a means to support and enhance nuclear security [9].

Security culture is important to be implemented and maintained by the licensee as senior management so that all potential threats can always be prevented and dealt with. The security culture must also be always communicated to all parties so everyone knows.

Based on NSS 7 - Nuclear Security Culture: managers influence culture throughout their organization through their leadership and management practices. With ongoing efforts, and by using the incentives and disincentives they have, they must establish behavior patterns and even change the physical environment. Senior managers are responsible for defining and revising safeguard policies and objectives; the operational manager is tasked with initiating practices that are consistent with this goal. Through their behavior, managers demonstrate their commitment to nuclear security and, as such, play an important role in promoting the culture of nuclear security within organizations.

Security-related Resource Management

Included in the management of resources related to security are:

1. Supply Chain Management (SCM) including procurement, purchasing, planning, and storage;
2. Facilities and infrastructure;
3. Human resource; dan
4. Funding.

Resource management is an important component in ensuring security is maintained and reliable. SCM keeps security processes and operations lean, ensuring that resource requirements are always available so that every necessary action can always be taken on time and the right size.

Facilities and infrastructure are needed so that security systems and activities can run in the best quality.

The right size of human resources and the competencies needed to make security activities can be carried out with the right procedures.

Adequate funding is also an important aspect so that security-related needs can always be met.

The licensee as the highest responsibility holder regarding security must ensure that the required resources are always available.

Operation Security

According to TDL-004, security operations include [10]:

Potential Threat Analysis

The licensee must be able to identify the potential threats that can occur to the facility as well as the things that can be done to overcome them. Examples of potential threats are theft and sabotage. Furthermore, the licensee must be able to do this to prevent such things as access restrictions, door locks or locks, bomb detection systems, and others.

Analysis of Target Potential

The licensee must be able to identify potential targets that will be targeted by the enemy in the facility and what can be done to overcome them. Examples of potential targets are nuclear material that has the potential to be a target for theft or illegal transfer.

Physical Security

The physical security system is carried out in line with the physical protection system. According to Article 4 paragraph (2) of the BCR Number 1 of 2009 concerning Provisions for the Physical Protection System of Nuclear Materials Installation and Material, the physical protection system aims to [11]:

1. Prevents unauthorized transfer of nuclear material;
2. Rediscovering lost nuclear material;
3. Prevent sabotage of nuclear installations and materials; and
4. Mitigate the consequences caused by sabotage.

Strategies and plans for physical protection in various nuclear power utilization facilities certainly vary depending on the level of threat that is acceptable. Furthermore, with different facility activities and layout so that there is a risk of individual threats, the licensee must identify the physical protection plan that needs to be carried out at each activity or point at the facility.

Physical protection strategies and plans also depend on the type of nuclear power installation or activity. The potential threat to research reactors and nuclear power plants is certainly different. Furthermore, the power that can be generated also affects the probability of an intrusion or a crime that can occur.

This is in line with the physical protection requirements required in NSS-13: Nuclear Security Recommendations on Physical Protection of Nuclear Materials and Nuclear Facilities, where "Physical protection requirements must be based on a graded approach, taking into account the evaluation of current threats, relative attractiveness, the nature of nuclear material and the potential consequences associated with the removal of nuclear material without permission and by sabotage of nuclear material or nuclear facilities." [12]

Personnel Security

There are two types of personnel who may be in the facility and are likely to commit crimes: employees and visitors.

The licensee must establish a security system to screen the personnel who will join the organization. Background checks of personnel before being recruited or employed in positions that interact with the security system need to be carried out to determine the level of trustworthiness. This is very important because based on the Inside Threat Report 2019 made by Haystax, as many as 70 percent of organizations say that insider attacks have increased in the last 12 months [13].

An examination of the background and purpose of visitors, including internship students, contractors, guests, and third parties is also important to know whether these parties have certain motivations for committing crimes in the facility.

Restrictions on access and escort must also be considered according to the potential level of threat and possible targets. The guards must also be carefully selected so that they are not easily distracted or defeated during the escort.

Information Security

Information security is needed to protect sensitive information that cannot be known by just anyone. One of the weaknesses in regulation regarding nuclear security in Indonesia is the lack of emphasis on information security and prioritizing physical security. In fact, a good crime strategy will first be done by gathering target information.

Information that needs to be protected includes information in print, digital, photo, audio, video, and knowledge. This information security must be adjusted according to the classification of the sensitivity level of the information and the impact if the information is known to an unwanted party.

Information security can be done among others by limiting access to information, information storage security systems, and screening of personnel who interact at information sources. Besides, communication security management is needed to keep the information unknown to unwanted parties [14].

Computer Security

Computer security is needed because in the industrial era 4.0 there will be a lot of use of network-based equipment, artificial intelligence, and automation. In addition to regulations regarding information security, cybersecurity is also a matter that is still not strong enough to be regulated in nuclear security regulations in Indonesia. According to the 2019 Honeynet Project Annual Report published by the National Siber and Coding Agency (BSSN), in the January-December 2019 range, Indonesia had 10,064,615 cyber attacks (the second highest in the world) [15]. Hence, computer security is necessary that the software used is not easily infiltrated by the enemy for protecting information is maintained and sabotage does not occur during operation. Restricting access to software and providing anti-virus is very necessary so that the software remains secure.

In addition, security components such as anti-virus and firewall security must be constantly updated to anticipate the development of sophisticated digital hacking systems.

Security Forces

Security forces are needed for security activities starting from the point of entry of the facility to security measures in the event of a crime in the facility. The security forces must have the competence and routinely refresh security-related competencies. The security unit must also be equipped with the equipment needed to carry out security activities and be armed if necessary.

The needs and planning of the security forces are adjusted to the level of potential threats that can occur at the facility.

Performance Appraisal and Potential Improvement

In the framework of evaluating performance, licensees must periodically carry out:

1. Internal audit;
2. Lesson learned;
3. Stakeholder assessments, including assessments of security-related leadership; and
4. Management review.

After obtaining a performance appraisal through the activities above, the licensee must determine the potential improvements that can be made including if necessary to change goals, objectives, organization, procurement of resources, changes in security systems, and others. The licensee must also determine a plan to implement the potential improvement that has been obtained.

CONCLUSION

From the results of the discussion, it can be concluded the importance of implementing a security management system in the nuclear field, given the increasing incidents of security, especially cybersecurity.

From the results of the literature study also found things that need to be regulated in the security management system document in the nuclear field, including:

1. Security objectives and how to achieve them;
2. Graded approach;
3. Leadership on Security;
4. Security culture;
5. Resources management;
6. Operation security (physical security, personnel security, information security, and computer security);
7. Security forces; and
8. Performance appraisal and improvement potential.

ACKNOWLEDGMENTS

The author would like to thank Mr. Bambang Eko Aryadi as the caretaker director of DP2IBN, Mr. Azizul Khakim as the supervisor in writing the paper, Mr. Zulfiandri for the support of the material and discussion, as well as DP2IBN personnels for their encouragement.

REFERENCES

[1] International Atomic Energy Agency, "IAEA Incident and Trafficking Database (ITDB): 2020 Fact Sheet," Vienna, 2020.

[2] Republik Indonesia, Peraturan Pemerintah Nomor 54 Tahun 2012 tentang Keselamatan dan Keamanan Instalasi Nuklir. Jakarta, 2012.

[3] International Atomic Energy Agency, IAEA Nuclear Security Series Glossary version 1.3, vol. 1.3, no. 2. 2015.

[4] Republik Indonesia, Peraturan Pemerintah Nomor 33 Tahun 2007 tentang Keselamatan Radiasi Pengion dan Keamanan Sumber Radioaktif. Indonesia, 2007.

[5] Republik Indonesia, Undang-Undang Nomor 10 Tahun 1997 tentang Ketenaganukliran, vol. 21, no. 3. Indonesia, 1997, pp. 295–316.

[6] British Safety Council, Annex SL The management system standard. London, 2015.

[7] International Atomic Energy Agency, General Safety Requirements Part 2 - Leadership and Management for Quality. Vienna, 2016.

[8] A. Fitriani, "LINGKUNGAN PENGENDALIAN INTERNAL TERHADAP KUALITAS LAPORAN KEUANGAN (Survei Pada SKPD Provinsi Sulawesi Tengah)," e J. Katalogis, vol. 5, no. 4, pp. 113–122, 2017.

[9] International Atomic Energy Agency, Nuclear Security Culture, no. 7. Vienna, 2008.

[10] International Atomic Energy Agency, Nuclear Security Management for Research Reactors and Related Facilities. Vienna, 2016.

[11] Badan Pengawas Tenaga Nuklir, Peraturan Kepala Badan Pengawas Tenaga Nuklir Nomor 1 Tahun 2009 tentang Ketentuan Sistem Proteksi Fisik Instalasi dan Bahan Nuklir. Indonesia, 2009.

[12] International Atomic Energy Agency, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities, no. 13. Vienna, 2011.

[13] Haystax, "Haystax Insider Threat Report 2019," Virginia, 2019.

[14] Advisera Expert Solution, "Clause-by-clause explanation of ISO 27001 White Paper." Zagreb, p. 26, 2016.

[15] Badan Siber dan Sandi Nasional, "Laporan Tahunan Honeynet Project 2019," Jakarta, 2019.

Study of Radiological Terrorism Pathways in The Region as a Reference to Create a Well-Targeted Policy on Combat Terrorism

Dwi Rahayu [1, a)]

[1]*Neutron scattering laboratory, National Nuclear Energy Agency of Indonesia*

[a)] dwi27@batan.go.id

Abstract. The most effective and efficient way to minimize the risk of nuclear terrorism is to prevent terrorist ambitions to get radioactive materials by locking up all radioactive materials to a gold standard through various physical and non-physical protections. Non-physical protection for example by making a prevention policy. Understanding the pathways that terrorists might be used to acquire radioactive material will certainly result in a well-targeted policy. This paper aims to analyze and discuss radiological terrorism pathways in the region based on incidents in the past. This study used a descriptive qualitative approach. The results of the study have shown there were seven pathways in which terrorists might be obtained radioactive material, namely insider, outsider, nuclear black market, robbery, state/organization, license fraud, and orphan sources. Cooperation between countries and/or the authorities is needed to minimize or even eliminate the possibility of radiological terrorism.

Keywords: Radiological terrorism, radioactive material, pathways

INTRODUCTION

In addition to nuclear disaster events such as in Chernobyl and Fukushima, we must also be aware of nuclear terrorism. Nuclear terrorism itself is interpreted by many scientists as an event with a small probability but with very large consequences that can affect the economic stability and security of the country[1].

Nuclear terrorism, according to Stanislav Ivanov, is related to power struggles, theft, illegal acquisition, movement, and use of other nuclear or radioactive materials with the intention of causing widespread damage to the population, economy, or environment to intimidate and pressure society and government[2]. Forms of nuclear terrorism also vary, for example, nuclear explosions originating from weapons or nuclear bombs, the release of radiation from radioactive material and the spread of radioactive material[1]. Other references also mention three types of nuclear terrorism, namely, the detonation of nuclear bombs, the sabotage of nuclear facilities which causes a massive release of radiation, and the use of radiological dispersion devices or dirty bombs to spread radioactive material and create panic and destruction[3].

Charles D. Ferguson and William C. Potter divide nuclear terrorism into four types: Theft and detonation of a complete nuclear weapon, Improvised Nuclear Device (IND), attacks and sabotage of nuclear installations, especially nuclear power plants, cause the release of radioactive material into the environment, and the acquisition of unauthorized radioactive material that contributes to the manufacture and detonation of Radiological Dispersion Device (RDD)—a “dirty bomb”—or Radiation Emission Device (RED)[4].

Between 2000 - 2010, Worldwide terrorism incidents were recorded 40129 incidents with the scope of the region are North America, the Middle East / Persian Gulf, Latin America, South Asia, Western Europe, East, and Central Asia, Eastern Europe, Southeast, Asia, and Oceana, Africa[5]. Although the data has not recorded any incidents of terrorism using radioactive agents, according to the fact sheet, the possibility, opportunity, and motive for the perpetrators to use radioactive substances in acts of terrorism still exist. The world can successfully prevent radiological/nuclear terrorism by only doing one thing, namely preventing terrorist ambitions to obtain radioactive materials by locking up all radioactive materials to a gold standard[6].

To make a gold standard, of course, requires an effort that is not easy, one of which is to understand terrorist minds and strategies to obtain radioactive sources, namely by understanding what the pathways may be used by terrorists.

In this paper, the author focus on discussing the pathways of terrorists get radioactive material for the type of radiological terrorist, RED, or RDD so that later the Government can make policies to prevent acts of nuclear terrorism that are directly related to the problems encountered. The region in this paper refer to the area that became a target of terrorism anywhere in the world, for example government office, crowded public facilities, critical infrastructure, etc.

The impacts that will arise when radiological terrorism occurs are health risks related to increased radiation in the environment, public fear, chaos, and economic crisis, etc. Therefore, it is hoped that this paper can have a positive impact on efforts to improve the performance of national nuclear security and/or its supervision so that the risk of nuclear terrorism can be minimized.

LITERATURE REVIEW

In the Oxford dictionary[19], terrorism is defined as "the unlawful use of violence and intimidation, especially against civilians, in the pursuit of political aims". The RAND Database of Worldwide Terrorism Incidents (RDWTI) recorded 40129 terrorism incidents during 2000-2010 that occurred in various parts of the world, as shown in **TABLE 1**.

TABLE 1. RAND Database of Worldwide Terrorism Incidents[5].

No	Region	Terrorism insidents
1	Middle East/Persian Gulf	14718
2	South Asia	5943
3	Southeast Asia & Oceania	2688
4	Western Europe	2616
5	Latin America	1663
6	Eastern Europe	1128
7	Africa	681
8	North America	118
9	East & Central Asia	103
10	Central Asia	4
11	Somalia	1
12	Southeast Asia and Oceania	1

Terrorists use a variety of weapons such as remote-detonated explosives, biological agents, fire or firebomb, chemical agents, explosives, radiological agents, firearms, knives, and sharp objects. In the database, RAND writes the use of a weapon radiological agent, although it has never happened in the world, the risk of using radioactive substances is probably not 0. Figure 1 shows the information on the weapons used by terrorists 2000 – 2010[5].

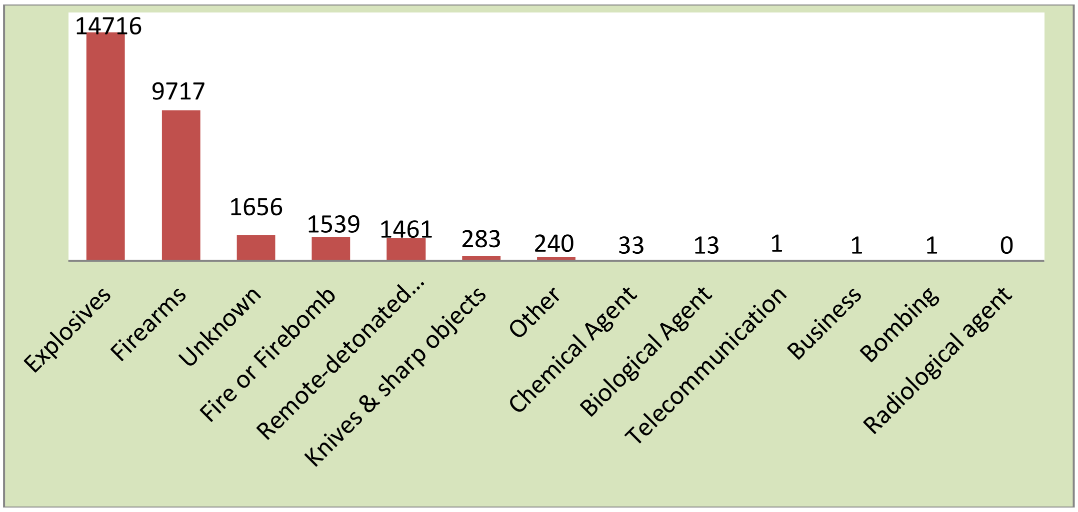


FIGURE 1. Weapons used by perpetrators in acts of terrorism (source: RDWTI)[5]

Nuclear terrorism is a series of criminal acts related to the forcible confiscation, illegal acquisition, and theft of nuclear weapons or radioactive materials to cause damage, public fear as an effort to intimidate the public and government [2], or in short, nuclear terrorism can be understood as acts of terrorism using nuclear weapons or radioactive materials. Many people are afraid of radioactivity due to ignorance of radiation and its effects. This condition is used by terrorists to spread fear, panic, and significant social disturbances [4] [18].

Radiological attacks occur in active and passive forms. Active radiological attacks through RDD or often called dirty bombs. A dirty bomb consists of a conventional explosive mixed with radioactive material [23]. Meanwhile, passive radiological attacks are carried out through RED, for example hiding radioactive substances with high levels of radioactivity in crowded places such as stations, airports, malls or deliberately contaminating food or drinks with radioactive materials [23].

Radioactive materials are widely available when compared to fissile weapon-grade materials, terrorists will likely use RDD and RED instead of using nuclear weapons and IND [4].

A report from the IAEA (International Atomic Energy Agency) in the IAEA Incident and Trafficking Database (ITDB) which consists of 139 countries stated that during 2019 there were 189 incidents reported by 36 member countries indicating unauthorized activities involving radioactive materials[20].

From 1993 to 31 December 2019 the IDTB contains 3686 confirmed incidents consisting of 290 incidents involving malicious use or trafficking, 1023 incidents whose information is not sufficient to be defined as malicious use or trafficking, and 2373 incidents that are not related to malicious use or trafficking[20].

Several incidents in the world that involve malicious use or trafficking, for example in 2008 Deutsche Welle reported that the security manager of a bank in Ukraine and an employee at the Ukrainian Embassy in Germany were arrested in Cherkassy. They carried the radioactive metal, which has an estimated price of 3.1 million euros. They transport uranium and cesium from Kiev in cars for sell to organized crime groups[21].

In 2006, the Belarusian prosecutor's office arrested a group of criminals who had planned to plant radioactive material at Internal Affairs Ministry offices in Kalinkavichy and Mazyr. Belarusian police also seized four containers containing radioactive material, firearms, grenades and explosives. Of the 20 gang members identified, 17 have been arrested and charged[22].

On August 16, 2009, a man was detained at the Dolbino checkpoint at Belgorod station while on the Nikolaev to Moscow train. This man carries 28 sets of radioactive night-vision devices that emitted over 600 times the background level of radiation[22].

Based on a review of this literature study shows that the problem of terrorism involving the use of radioactive materials is experienced and discussed by countries in the world as a problem that requires special attention. Cooperation between countries is needed to minimize or even eliminate the possibility of smuggling, illicit trafficking, and theft of radioactive material.

METHODOLOGY

This study used a descriptive qualitative approach, which means that this study is a literature study using books, scientific research articles, newspapers, reports, and other literature as the source and/or the main object of the study. The information that has been collected then carried out a descriptive analysis which provides a clear, objective, systematic, analytical, and critical description of the Radiological Terrorism Pathways so that a comprehensive summarization about it can be obtained[25].

RESULTS AND DISCUSSION

The cycle of nuclear materials both used in the nuclear industry and nuclear research institutes are shown in **FIGURE 1**.

At every level of radioactive material users, there is always a special permit/license from the regulator, from the production process to the radioactive waste material management. The first stage of the cycle of the radioactive sources, is the production process, one of the places where the production of radioactive material is nuclear reactors. After the radioactive source is obtained, the next step is the manufacturing process for various purposes and uses. An unsealed radioactive source is usually used in the health sector and a sealed radioactive source is used for industry, health, etc. After this manufacturing process must be tightly controlled where the radioactive source is used, do not let illegitimate users get this radioactive source[4].

After a certain period of time, the activity of radioactive source will run out depending on the half-life of the radioactive source. Then the radioactive source becomes a material that is not used anymore. The next stage is to allocate the radioactive source to the government waste treatment institution or return it to the producer where the radioactive source was purchased. These disused sources also still need good, thorough, and careful handling. If the users consider that disused sources are no longer economical, its thrown away in the environment then becomes orphan sources, then this is very dangerous[4].

Every step in the life-cycle of a radioactive source requires well supervision and security so that terrorists cannot acquire it. The following chapters will discuss the pathways of radiological terrorism based on past incidents.

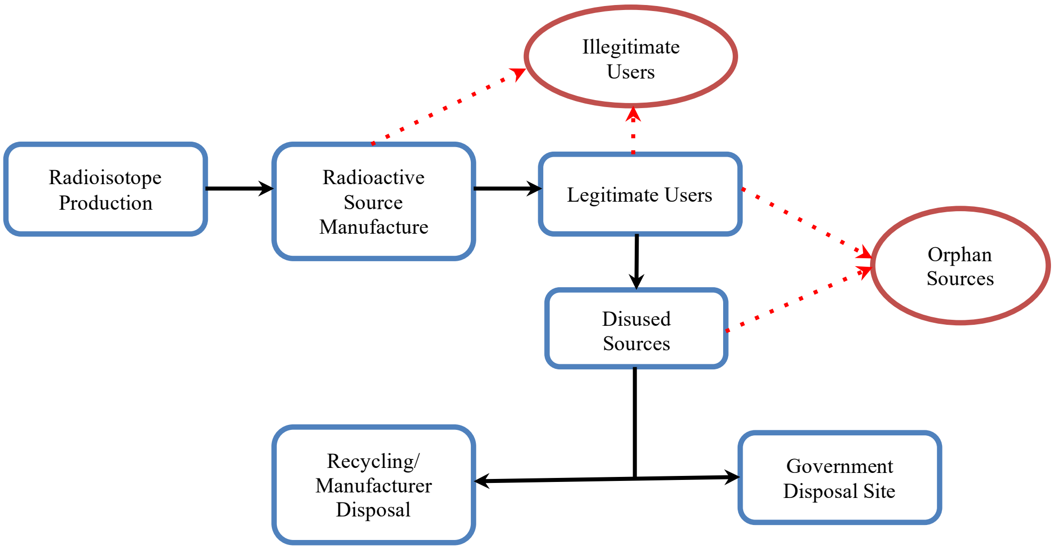


FIGURE 2. The life-cycle of a radioactive source [4]

Pathways of Radiological Terrorism

Insider

The attack can be carried out by using insiders, that is by tricking them into pretending to be legal or promising to give rewards for doing what they are told. Insiders can consciously crimes for personal gain, but can also unconsciously. The person is only doing what was ordered without understanding what he was doing. By using insiders, it tends to be easier because it is recognized as a legal person so that they have access to anywhere so that they are free from security checks. Because of this, this method has a high success rate. Unfortunately, crimes with insiders are very difficult to overcome. Must have enough evidence, and a system that supports the handling program of the insider[3][4].

The criteria that determine the success or failure of the crime of using insiders are: screening to ensure that employees can be trusted when and outside of work; corruption, collusion, and nepotism practices in the organization; some employees can be influenced and have no integrity; the financial condition of employees so that it will encourage to commit a crime; have relations with terrorist groups;[3].

Several incidents involving insider at nuclear facilities in the world:

1. Russia, Glazov, Chepetsky Metallurgical Plant (1991 - 1992)

At least 12 people were involved in the theft of several kilograms of Low Enriched Uranium (LEU). The theft was carried out by diverting 4% of the material as "Inventory loss that allowed" for several months. After the material is collected, the perpetrators try to smuggle to another location. After an in-depth investigation, there was 300KG LEU missing from the facility[9].

2. Lithuania, Visaginas, Ignalina Nuclear Power Plant,

In 1992, there was a theft of a 7m-long fuel assembly, which contained LEU nuclear fuel of around 100 Kg. The perpetrators were employees and the nuclear power plant guard team. The radioactive material is successfully smuggled out of the facility using the bus[9].

3. Poland, Borne Sulinowo, Borne-Sulinowo garrison (1992)

A report in 1992 stated that two containers of Cs-137 had been stolen from the Soviet Borne-Sulinowo garrison. Perpetrators are soldiers assigned to guard the place and two local farmers. After 12 days of searching, it was found one container containing 88 kg of material and a second container containing 35 kg of unknown material[9].

Outsider

Outsider is defined as an attack carried out by people outside a facility that is not related to the organization of the nuclear facility. Should be aware of the possibility that the outsider will cooperate with the insider, of course, this will make the success rate even higher.

Some examples of outsider attacks on nuclear facilities in the world:

1. U.S.A., Pennsylvania, Three Mile Island Nuclear Power Plant

In 1993, there was an outsider attack on the Three Mile Island Nuclear Power Plant. The culprit named Pierce Nye, who was known to have just left the mental ward. He drove his station wagon through the outer gate which opened at the turn of the guard at 7 am. He managed to enter the turbine area and get around for 4 hours. Although this action did not cause theft and or access to vital areas of the plant, this was an event that could be used as a lesson so that unauthorized personnel could not enter the nuclear facilities[9].

2. Russia, Nizhny Novgorod Oblast, The closed city of Sarov (1994)

Three teenagers reportedly stole 9.5 kg of U-238 from the closed city of Sarov, Russia's nuclear research center. They allegedly will sell nuclear material to get money to buy video equipment[9].

Nuclear Black Market

Advances in payment technology using virtual currency and highly sophisticated IT technologies make money laundering difficult to track. Transactions on the black market are the concern of many countries because many criminals do their actions here.

A book written by Charles D. Ferguson and William C. Potter [4] explains that trying to buy weapons or nuclear material on the nuclear black market seems to be a very possible choice for terrorist groups looking for nuclear weapons. Not only radioactive material that may be traded but the technology as well. Stephen Herzog (2020)[7] states that during the 1980s, Libyan leaders obtained technology to build nuclear weapons from the proliferation of black markets. He has ambitions to make nuclear weapons using both uranium and plutonium. AQ Khan and his proliferation network allegedly supplied gas centrifuge equipment to launch that ambitions for making nuclear weapons[4][7].

Marauder/Robber

Marauder and Robber have a slightly different meaning. A marauder is a person who goes around a place in search of things to steal or people to attack, while a robber is a person who steals from a person or place, especially using violence or threats.

In this case, there are two possibilities for terrorists to get radioactive material, first, robbery during the war, political unrest and chaos, and second, robbing or stealing radioactive material during transportation. For example, nuclear facilities in Iraq were damaged or destroyed effectively by looting, which began in early April 2003. Robbers stole stationery and other equipment, as well as barrels that once contained low-enriched uranium. This action raises concerns that around 200 radioactive sources on Tuvait may have been stolen. Some sources might be strong enough to trigger a dirty bomb[4].

Examples of incidents involving marauders against radioactive material occurred in Mexico. On 6 December 2013, it was reported that six people were arrested in connection with the theft of trucks with radioactive waste. A truck with cobalt 60 was taken from a gas station where the driver stopped. Material, hospital waste that is transported from Tijuana to a warehouse near Mexico City, is often referred to as potential material in a dirty bomb, in the form of a combination of explosives and radioactive materials[10].

On August 3, 2017, Independent.co.ug and sputniknews.com reported that thieves stole a nuclear densitometer from the back of a truck in a parking lot in a shopping center in the northern city of Monterrey while the driver was eating at a fast-food restaurant. The device used to measure the density of land belongs to the civil engineering department at the University of Nuevo Leon and contains nuclear material that is "very dangerous" if not handled properly[11][12].

License Fraud

License fraud can happen in the process of importing or exporting radioactive sources. Although regulators explicitly require licenses in the radioactive buying and selling process, buyers and sellers are obliged to exchange information about the licenses before sending sources, buyers can try to place fake licenses without the seller knowing. When the seller feels that the buyer's license is legal, then the seller has no reason not to send the radioactive sources.

Examples of license fraud events in the world: The first case involved Stuart Lee Adelman, also known as Stuart von Adelman, who for several years from the 1980s to the 1990s obtained radioactive material through illegal licenses, including license fraud. In 1996, he was arrested in the United States, where he pleaded guilty to a federal crime for illegally obtaining radioactive material[4]. In the second case, in May 2003, an official in Argentina investigated what was originally a suspicious request from someone from Texas to send cobalt-60 for use in a therapy machine. "Licenses" arouse suspicion because the written license is used for dental x-ray machines. Although the FBI investigation into the incident reportedly did not reveal terrorist activity, this indicated the need for better import and export controls, so license fraud would not occur in the future[4].

State / Organization

Although there is no evidence that countries intentionally transfer radioactive material to terrorist groups, Matthew Bunn said that the decision to buy radioactive substances could be initiated by certain countries or organizations, and not by a group. The state can transfer these materials to terrorist organizations, or government officials who have access to these materials can transfer them to terrorists for ideological reasons or mercenaries[4][8].

Orphan Sources

IAEA states that “orphan sources are radioactive sources that have sufficient radiological hazards to ensure regulatory control, but which are not under regulatory control because they have never been, or have been abandoned, lost, stolen or moved without proper permission”. Vulnerable sources are those currently under the control of the regulatory body, but the level of control is weak[13].

That can be seen as a source that can easily become orphans. In recent years, the source of orphans has caused many deaths or serious injuries when people find it. This problem, as well as concerns that orphaned or vulnerable sources can be obtained for malicious purposes, have prompted many countries to consider taking collective action to increase their control[13].

There are three sources for orphans: import sources of orphans, refusal to control sources that were controlled in the past, sources that were not controlled in the past, including domestic NORM sources (Naturally Occurring Radioactive Material)[14].

The main stages of source life are production, import, export, transportation, distribution, use of resources, maintenance and repair, processing, demolition and storage. The regulator must control the source at all the stages mentioned, using a national source register system that is related to the registration and licensing system, as well as with the control system[14].

To get sources of orphans, terrorists can use radiation detectors or look for other signs, such as heat emissions[4]. For example, the cases that involving orphan sources are: In February 2000, a serious accident in Samut Prakarn, Thailand, caused widespread death, injury and anxiety. The missing source of cobalt-60 teletherapy is stored, apparently without the knowledge or permission of the regulatory agency, in an unsafe outside area that is usually used to store new cars. Two local scrap collectors allegedly bought the memo, including its source, and brought it home to dismantle it and resell it. Then, they threw the head of the teletherapy partially dismantled into a landfill, resulting in three deaths[15]. In late December 2001, in the Republic of Georgia, three loggers discovered two powerful strontium 90 sources, which were abandoned and used to power radioisotope thermoelectric generators. These people, who don't have radiation detectors, receive strong radiation, trying to stay warm from warm objects[24]. Another example is an incident that occurred in 2008, Karachi, Pakistan. Around the source of oil and gas, an orphan source was found in the form of two containers, which should have been abandoned after Soviet oil drilling operations in the late 1960s[16].

CONCLUSION

Based on the studies that have been conducted, it is known that the opportunity of terrorists will use radioactive materials for their actions in the future is still exist. Therefore, incidents in the world involving malicious use or trafficking of radioactive material became a special concern in many countries. The results of the study have shown there were seven pathways in which terrorists might be obtained radioactive material, namely insider, outsider, nuclear black market, robbery, state/organization, license fraud, and orphan sources. It is hoped that this paper can have a positive impact on efforts to improve the performance of national nuclear security and/or its supervision so that the risk of nuclear terrorism can be minimized.

ACKNOWLEDGMENTS

My sincere gratitude goes out to my husband for his remarkable professionalism and feedback during a difficult period. Additionally, I would like to thank Ksenofontov Alexander Ivanovich, Kosterev Vladimir Viktorovich, Demin Viktor Maksimovich and two anonymous peer reviewers who will remain unnamed for comments and/or encouragement while completing this study.

REFERENCES

[1] Phineas Dickstein and Sarah Vanunu, *Nuclear Terror: The Essentials, Threats, Effects and Resilience. The International Institute for Counter-Terrorism (ICT),* 2016, p. 9-12, 36.

[2] Stanislav Mikhailovich Ivanov, *Предотвращение угрозы ядерного терроризма – веление времени,* available at https://www.belfercenter.org/sites/default/files/legacy/files/ivanov-preventing-nuclear.pdf

[3] Matthew Bunn, Martin B. Malin, Nickolas Roth, and William H. Tobey, *Preventing Nuclear Terrorism Continuous Improvement or Dangerous Decline?* (Cambridge, MA: Project on Managing the Atom), Belfer Center for Science and International Affairs, Harvard Kennedy School, 2016, p. 24.

[4] Charles D. Ferguson and William C. Potter, with Amy Sands, Leonard S. Spector, and Fred L. Wehling, *The Four Faces of Nuclear Terrorism,* Center for Nonproliferation Studies, Monterey Institute of International Studies, 2004, p. 13, 200, 269 – 293.

[5] *"RAND Database of Worldwide Terrorism Incidents."* Available at http://smapp.rand.org/rwtid/search.php

[6] Allison, Graham. 2010 *"Nuclear Terrorism Fact Sheet."* Policy Memo, Belfer Center for Science and International Affairs, Harvard Kennedy School.

[7] Stephen Herzog, *The Nuclear Fuel Cycle and the Proliferation "Danger Zone"*, Journal for Peace and Nuclear Disarmament, 3:1, 60-86, DOI:10.1080/25751654.2020.1766164, p.11

[8] Matthew Bunn, Guardians at the Gates of Hell: Estimating the Risk of Nuclear Theft and Terrorism and Identifying the Highest-Priority Risks of Nuclear Theft, Massachusetts Institute of Technology, 2007, p. 111-159.

[9] Nuclear Facilities Attack Database (NuFAD),2020, available at http://www.start.umd.edu/nuclear-facilities-attack-database-nufad

[10] By Randal C. Archibold and Paulina Villegas, 2013, *6 Arrested in Theft of Truck With Radioactive Waste*, available at https://www.nytimes.com/2013/12/07/world/americas/mexico-radiation-exposure.html

[11] Sputnik, 2017, Mexican Authorities Find Stolen Container With Radioactive Materials - Official, available at https://sputniknews.com/latam/201708031056137498-mexico-radioactive-materials-theft/

[12] The Independent, 2017, Thieves steal radioactive material in Mexico again, available at https://www.independent.co.ug/thieves-steal-radioactive-material-mexico/

[13] IAEA*, Strengthening control over radioactive sources in authorized use and regaining control over orphan sources,* IAEA-TECDOC-1388, 2004, p. 3.

[14] Helena Janžekovi, Janez ešarek. Orphan Sources in Slovenia. International Conference Nuclear Energy for New Europe. Slovenian Nuclear Safety Administration. 2005, p. 2.

[15] IAEA, "The Radiological Accident in Samut Prakarn", 2002, available at https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1124_scr.pdf

[16] Baqir Sajjad Syed, 2008, *Containers found with radioactive material*, available at https://www.dawn.com/news/311285/containers-found-with-radioactive-material

[17] Allison, Graham. 2010 *"Nuclear Terrorism Fact Sheet."* Policy Memo, Belfer Center for Science and International Affairs, Harvard Kennedy School.

[18] Dr. Frank Barnaby. *"Dirty Bombs and Primitive Nuclear Weapons"*. Oxford research group. 2005, p.1. Availeble at http://www.oxfordresearchgroup.org.uk/sites/default/files/dirtybombs.pdf

[19] "Definition of Terrorism", Available at https://www.oxfordlearnersdictionaries.com/definition/english/terrorism?q=terrorism+ ;https://www.lexico.com/definition/terrorism

[20] IAEA INCIDENT AND TRAFFICKING DATABASE (ITDB), "Incidents of nuclear and other radioactive material out of regulatory control 2020 Fact Sheet", Available at https://www.iaea.org/sites/default/files/20/02/itdb-factsheet-2020.pdf

[21] DPA news agency (kjb) , 2008, available at https://www.dw.com/en/ukraine-embassy-worker-arrested-for-radioactive-smuggling/a-3467700.

[22] Alex P. Schmid & Charlotte Spencer-Smith, 2012, *"lllicit Radiological and Nuclear Trafficking, Smuggling and Security Incidents in the Black Sea Region since the Fall of the Iron Curtain– an Open Source Inventory*", PERSPECTIVES ON TERRORISM Volume 6, Issue 2, p. 24.

[23] Steinhäusler Friedrich, Rydell Stan, and Zaitseva Lyudmila. "*Risk Due to Radiological Terror Attacks With Natural Radionuclides"*. American Institute of Physics. 2008, p. 3 - 5.

[24] Prof. G. Nabakhtiani, L. Chelidze, "Recovery of Orphan Radioactive Sources in Georgia (Lessons Learnt)", availeble at https://www.eu-alara.net/images/stories/pdf/program12/Session3/s3-4-chelidze.pdf

[25] Carrie Williams, "Research Methods", Journal of Business & Economic Research – March 2007 Volume 5, Number 3, p 1 - 8.

The Readiness to Implement Global Nuclear Security Regime and Physical Protection System Readiness for Supporting Commercial-Scaled NPP in Indonesia

Khairul Khairul[1,a)], Tegung Asmoro[1)], Fadhlur Rahman Al Faruqi [2)] and Anhar R. Antariksawan[3)]

[1)]*Information Utilization Center and Strategic Area-BATAN*
[2)]*Gadjah Mada University*
[3)]*Accelerator Technology and Science Center, BATAN*

[a)]*Corresponding author: khairul@batan.go.id*

Abstract. The International Atomic Energy Agency (IAEA) as an international organization dealing with the use of nuclear energy has an important role in ensuring efficiency, responsibility, and sustainability of the development of nuclear energy. The international organization already provides nuclear security regulations and recommendations both legally binding and non-legally binding as a guideline for member states. Indonesia as one of the IAEA members must implement nuclear security, physical protection system, and nuclear security culture to ensure the safe and secure use of nuclear energy. This paper aims to assess the readiness of the implementation of a global nuclear security regime and physical protection system at a nuclear facility in anticipation of embarking a commercial-scale nuclear power plant (NPP) development in Indonesia. The assessment is reviewed from the aspect of legal instruments, the application of the physical protection system, and the implementation of nuclear security culture. From the study, it is known that Indonesia has prepared various regulations and guidelines related to nuclear security, adequate experience in the application of a physical protection system at the existing nuclear facilities, and good implementation in nuclear security culture. In general, Indonesia is ready to implement a global nuclear security regime and physical protection system to anticipate the development of a commercial-scale nuclear power plant in Indonesia.

Keyword. global nuclear security regime, physical protection system, nuclear security culture, nuclear power plan.

INTRODUCTION

The idea of building a commercial-scale Nuclear Power Plant (NPP) in Indonesia is based on the consideration that fossil energy sources which have been the main pillar in electricity generation in Indonesia are running low. Electricity demand from various sectors has increased by around 7% per year [1]. It will be difficult to fulfill the demand if energy sources only rely on fossil fuels, especially if it relates to the environmental carrying capacity of air pollutants. Electricity demands and clean environment quality are requirements that must be met in electricity generation in Indonesia in the future.

Nuclear security and physical protection are one of the 19 aspects of infrastructure development in NPP, as stated in IAEA guidelines No. 1358 Evaluation of the Status of National Nuclear Infrastructure Development [2]. Nuclear security is an effort to prevent, detect, and respond to criminal or malicious acts involving or directed at nuclear materials, other radioactive materials, related facilities, or related activities. Other actions determined by the state to prevent adverse impacts on security must be carried out appropriately.

The implementation of nuclear security in each IAEA member state is very different. It depends on the Design Basis Threat (DBT) made by the authorities in each state because every member state has a different type of threat. DBT guidelines provided by the competent authority will be referred by each holder/operator in implementing a physical protection system (PPS) for a research reactor or power reactor [3]. IAEA cannot provide nuclear security standards guidelines as applies to nuclear safety through IAEA Basic Safety Standard documents like the IAEA Safety Standard Series because the implementation of nuclear security differs for each country. As a guideline for implementing nuclear security, IAEA provides a Nuclear Security Series (NSS) publication to help member states develop effective national nuclear security regimes. NSS guideline hierarchy, starting from the publication of Fundamental, Recommendation, Implementation Guide, Technical Guidance. Since 2006, IAEA began issuing NSS, until today IAEA has provided 39 series nuclear security guidelines, which are non-legally binding. However, some IAEA nuclear security guidelines were adopted by the regulatory agency into national regulations. Likewise, in Indonesia, Nuclear Energy Regulatory Agency (BAPETEN) adopted some guidelines as a legally binding reference for each holder/operator in the use of nuclear power and radioactive sources. The publication provided by IAEA complements international legal instruments on nuclear security, such as the Convention on the Physical Protection of Nuclear Materials (CPPNM) and its 2005 Amendments [4], the International Convention for the Suppression of Acts of Nuclear Terrorism 2004 [5], United Nations Security Council Resolutions 1373 [6], 1540 (UNSCR) [7], and Code of Conduct on the Safety and Security of Radioactive Sources, 2004 [8].

This paper aims to assess the implementation of the global nuclear security regime and PPS, moreover assess Indonesia's readiness to implement these in commercial-scale NPP. It is important for the public to understand that the security aspect is considered in commercial-scale NPP development.

MAIN SUBJECT DISCUSSION

Global Nuclear Security Regime

The main components of the global nuclear security regime as mentioned above are CPPNM and its amendments, The International Convention for the Suppression of Acts of Nuclear Terrorism (ICSANT), UN Security Council Resolution 1540, and IAEA activities and documents. CPPNM have been ratified by Indonesia into Presidential Regulation number 46/2009 on ratification of Amendment to the Convention on the Physical Protection of Nuclear Material [9] and ICSANT ratified by Indonesia into Presidential Regulation number 10/2004. The global nuclear security regime is a framework to achieve worldwide nuclear security implementation in a nuclear facility [10]. The global nuclear security regime becomes the reference, guideline, and framework for the state in implementing nuclear security. IAEA provides services in the form of IAEA Nuclear Security Services Program, as well as establishing International Physical Protection Advisory Services (IPPAS).

IPPAS was formed by the IAEA in 1995 to provide peer review on the implementation of international instruments and IAEA guidelines on the physical protection of nuclear materials and other radioactive sources, facilities, and other related activities [21]. The IPPAS mission consists of a national-level review of the rules and regulatory framework. Depending on a state's request, IPPAS can also review the PPS at nuclear facilities and the transport of nuclear and other radioactive materials. Cyber-security is also included in IPPAS missions if required by the permitted holder. The report from IPPAS Mission consists of recommendations and advice as well as good practices in research reactors that can be applied by member states in implementing the global nuclear security regime and PPS. The IPPAS team is assisting IAEA member states through an advisory on the implementation of a PPS that refers to the IAEA-Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Rev.5/2011). It strengthens the national nuclear security regime, PPS, and the necessary steps. Findings during a visit to nuclear facilities will be reflected in IPPAS mission reports that are given to member states. IPPAS mission reports are treated as highly classified documents by IAEA. Five areas will be reviewed by the IPPAS team, in which 1) National Level (Regulatory Framework); 2) Facility Level (Application of Physical Protection and Security Culture); 3) Security of Radioactive Sources; 4) Security During Transportation; 5) Cyber-Security. Apart from that, the IPPAS can follow up on the assistance provided by IAEA if a request is formally submitted by a member state. Assistance including training, technical support, and more targeted assessments of various elements of the national nuclear security regime is provided. IAEA Nuclear Security Services (IPPAS Mission) is done once in every 5 years.

Besides IPPAS, IAEA also provides technical cooperation in strengthening nuclear security, such as physical protection systems, security during transport of nuclear and radioactive material, cyber-security, and nuclear security culture. Indonesia as an IAEA member state has a lot of cooperation with IAEA in the field of nuclear security. In addition to IAEA, Indonesia also cooperates bilaterally with various countries to increase capacity in nuclear security for networking because nuclear security is cross-border activity. This will also strengthen the global nuclear security regime.

Physical Protection System in Indonesia

Based on paragraph 7 (section 1) BAPETEN Chairman Regulation No. 1 of 2009 states that every permitted holder and area manager must submit the PPS in the form of a confidential physical protection plan document to BAPETEN to get approval based on its requirements [11]. The physical protection plan document is always revised periodically and contains a complete description of nuclear materials in nuclear installations and their locations. The physical protection plan document also described the vulnerability analysis, threat scenarios such as theft and sabotage both conducted by outsiders and from insiders who are in nuclear facilities [12]. Physical protection organizations are also included in the physical protection plan, which explains the duties, responsibilities, authorities, and qualifications of each personnel. The training program of physical protection personnel is described in the document. Threat and target identification are linked to the categorization of nuclear materials based on safeguards issued by the regulatory agency and shown in Table 1(BAPETEN Chairman Regulation. Number 1 Year 2009). Based on these, the permit holder/operator of a commercial-scale NPP is required to prepare a physical protection plan and submit it to the regulatory agency for review. If the physical protection plan complies with the design basis threat, the regulatory agency will approve the physical protection plan document proposed by the permitted holder.

TABLE 1. Categorization of Nuclear Material

Material	Form	Category			
		I	II	III	IV
1. Plutonium	Unirradiated or irradiated with exposure ≤ 1 Gy/hr (100 rad/hr) at 1 m unshielded	≥ 2 kg	500 g < Pu < 2 kg	15 g < Pu ≤ 500 g	1 g< Pu ≤ 15 g
2. Uranium-235	Unirradiated or irradiated with exposure ≤ 1 Gy/hr (100 rad/hr) at 1 m unshielded				
	− Uranium enriched ≥ 20% U-235	≥ 5 kg	1 kg < U < 5 kg	15 g < U ≤ 1 kg	1 g < U ≤ 15 g
	− Uranium enriched between 10% - 20% U-235	−	≥ 10 kg	1 kg < U < 10 kg	1 g< U ≤ 1 kg
	− Uranium enriched above natural, but less than 10%U-235	−	−	≥ 10 kg	1 g < U < 10 kg
3. Uranium-233	Unirradiated or irradiated with exposure ≤ 1 Gy/hr (100 rad/hr) at 1 m unshielded	≥ 2 kg	500 g < U < 2 kg	15 g < U ≤ 500 g	1 g< U ≤ 15 g
4. U-natural, U-depleted, Th and bulk nuclear material waste	Unirradiated or irradiated with exposure ≤ 1 Gy/hr (100 rad/hr) at 1 m unshielded	−	−	≥ 500 kg	1kg < U/Th < 500 kg
5. Irradiated Fuel (U-natural, U-depleted, Th or enriched fuel<10 %)	- for transportation	--	Unlimited	--	--
	- for storage/use	--	--	Unlimited	--

Vital equipment in the protected area also described in the document. If vital equipment in the nuclear installation is sabotaged, it can result in the release of radioactive substances and cause contamination to the environment. Permitted holders must also be able to estimate the consequences and response teams.

Program for Supporting Physical Protection System

PPS needs support in the form of (1) establishing a maintenance program, (2) contingency plans, (3) security culture, (4) confidentiality of information. PPS maintenance program, functional testing, and maintenance program for physical protection devices to ensure product quality has been carried out by the PPS maintenance team on PPS's essential elements to improve tool capabilities and system quality. These elements are tested to ensure tool reliability and tool maintenance programs, which are effective for controlling and maintaining equipment used in nuclear facilities.

After establishing a maintenance program, another supporting element of the PPS is the establishment of contingency plans. It is a series of systematic and planned activities carried out to anticipate emergencies caused by threats in the form of security breach such as theft and sabotage, or other malevolent human attacks of nuclear installations and/or material, and/or during transportation of nuclear materials. Contingency planning is an important part of the ability of a PPS at the facility to deal with a problem that comes from the security aspect effectively and successfully.

The third element is the security culture. The application of security culture is an obligation for every manager of nuclear materials as the fundamental principle of physical protection. Security culture is part of the nuclear security program that applies to all fields, sections, divisions, and units that have an important role.

Nuclear security culture can enhance the effective implementation of nuclear security in facilities and transportation. Nuclear security culture defines human factors as an asset in a nuclear security program. Strong nuclear security culture will provide a great guarantee that the entire nuclear security system will be able to prevent, detect, delay, and respond to theft, sabotage, unauthorized access, illegal transfer or other malicious actions involving nuclear and radioactive material related to facilities and transportation [13]. IAEA has provided recommendations and guidance on carrying out a self-assessment to measure the effectiveness of the application of nuclear security culture. Indonesia is currently known as the first IAEA member state to carry out a self-assessment of nuclear security culture at the research reactor facility[14].

The fourth element is the confidentiality of the information. Nuclear security culture encourages awareness about the sensitive nature of nuclear security information and the necessity to protect the confidentiality of this information. The information should not be shared in public because it can be used for malicious purposes. Therefore, BATAN must establish criteria for determining sensitive information in the nuclear security field.

METHODOLOGY

The method in writing this paper is to conduct a literature study on nuclear security, specifically international treaties related to nuclear security, binding, and non-binding global and national nuclear security recommendations, as outlined in the subject. Furthermore, to assess Indonesia's readiness in terms of nuclear security, a discussion and study of the adoption of treaties and recommendations for nuclear security at nuclear facilities in Indonesia, particularly in BATAN, will be carried out. Based on this, it can be concluded Indonesia's readiness in implementing the global security regime and PPS for commercial-scale NPP. In this case, extrapolation from the implementation of nuclear security at NPP facilities does not fully describe all aspects, but its basic aspects (such as in the case of nuclear security culture) are taken into consideration because in principle nuclear materials which were managed at the BATAN nuclear facility are identical to NPP except in terms of its categorization of nuclear materials.

RESULTS AND DISCUSSION

Indonesia has shown its commitment in nuclear security implementation, especially on physical protection system and radioactive resources security that are related to the global nuclear security regime legal binding document and IAEA guides and its recommendations. From the law and regulation instrument aspects such as:

1. Indonesia CPPNM Amendment 2005 to Presidential Regulation Number 46 Year 2009 about Amendment to the Convention on the Physical Protection of Nuclear Material legalization [9]
2. Government Regulation of The Republic of Indonesia Number 54 Year 2012 on Nuclear Installation Safety and Security[15],
3. Government Regulation of The Republic of Indonesia Number 43 Year 2006 about Nuclear Reactor Permit [16],
4. Government Regulation of The Republic of Indonesia Number 33 Year 2007 about Radioactive Resources Ionization Security and Radioactive Source Security [17].
5. BAPETEN Chairman Regulation No. 1 Year 2009 (source: IAEA INFCIRC/225/rev.4, 2009).
6. BAPETEN Chairman Regulation No. 6 Year 2015 (source: IAEA NSS No. 11, Security of Radioactive Source).

All mentioned legal instruments and regulations are a part of national nuclear security regime and are accordance with the legal instruments in international level. These law instruments and regulation are crucial to ensure national and international stakeholders to have the level of trust that Indonesia has a strong commitment in nuclear security aspects. Those are also to show that Indonesia is committed to establish commercial-scaled NPP only for peace and solemnly protect the existing facilities at NPP from parties who want to cause disruption and threat to endanger community, both national and neighboring countries of Indonesia.

On the other hand, to support global nuclear security regime, Indonesia as the member country participates actively in supporting IAEA nuclear security activities, such as attended in every Broad of Governor (BOG) and yearly General Conference (GC) meetings of IAEA. Other than yearly meetings, the representative of Indonesia also actively involved in nuclear security agenda such as training, workshop, expert meeting and conference about nuclear security presented by IAEA also the regional and international organizations. Indonesia is also active in Nuclear Security Summit event from 2010 (USA), 2012 (South of Korea), 2014 (Netherland) and in 2016 (USA).

However, despite the availability of regulatory infrastructure, Indonesia still faces challenges because the national nuclear security regime has not been implemented properly by all stakeholders. This is reflected in the national nuclear security regime entity that does not fully understand the duties and roles related to nuclear security. National nuclear security regime should be the synergy between Coordinating Ministry for Political, Legal, and Security Affairs (Kemenko Polhukam-RI), National Police (POLRI), National Intelligence Agency (BIN), Military, Custom, National Nuclear Energy Agency (BATAN), and Nuclear Energy Regulatory Agency (BAPETEN), Maritime Security Agency (BAKAMLA). National level coordination followed by the all entities is not institutionalized well yet.

In technical SPF aspects, IPPAS mission study result is as a very important reference for a country to see the compliance level in implementing physical protection principal on nuclear installation and activities using nuclear material. IPPAS has visited Indonesia upon request from Indonesian Government, via BAPETEN,

1. IPPAS Mission: implemented in 2001;

2. IPPAS Mission: implemented in 2007;
3. IPPAS Mission: implemented in 2014;

These three missions are implemented at three BATAN research reactor locations (Bandung, Yogyakarta, Serpong) and at private radiation facilities. IAEA IPPAS team gave some recommendations, advices and good practice assessment that are implemented by BATAN.

With three times PPS evaluation by IPPAS team in Indonesia, some of the PPS implementation recommendations are not aligned with INFCIRC/225/rev.5 [18]. Because IPPAS report related to nuclear security is very sensitive, the information will not be described in this paper.

Referring to the IPPAS mission report, BATAN has made some efforts to meet the IPPAS recommendation and suggestion, discussing the physical protection implementation at nuclear facility in Bandung, Yogyakarta and Serpong according to categorization of nuclear material owned by each nuclear facility. Moreover, BATAN also has upgraded physical protection system as stipulated at (GR 54 Year 2012, Article 50-55), implementation of nuclear security culture and self-assessment of security culture [19]. Right now, BATAN still developing human reliability program implementation, as required in the Government Regulation No. 54 Year 2012, about Nuclear Installation Safety and Security (Article 64) and in BAPETEN Chairman Regulation No. 1 Year 2009, Nuclear Material and Facility Physical Protection Requirements.

In effort to respond IPPAS mission findings, in nuclear reactor facility level, BATAN has escalate the nuclear security capacity building through Gap Analysis Workshop referring to IAEA Recommendation-INFCIRC/225/rev.5/2011, that was performed at Serpong Nuclear Area (KNS), on September 9th to 12th 2014. In addition to that, Workshop on Performance Testing for Physical Protection based on INFCIRC/225/rev.5/2011 has been performed in the existing nuclear facility. These trainings are conducted by BATAN in cooperation with National Nuclear Security Administration-U.S. *Department of Energy (NNSA/U.S.DOE) in 2015 - 2016.* All physical protection trainings have been performed by BATAN in effort to gain knowledge from U.S. DOE's experts that has the experience to operate 98 NPP in the United States, (*sources: U.S.NRC, as of August 22th 2019*).

Another effort that has been made by Indonesia Government in regard to nuclear security is on April 2018, BAPETEN has held awareness meeting in Jakarta, attended by Director of IAEA Division of Nuclear Security and attended by attentive audiences from national nuclear security regime entities. One of the results of the meeting was National nuclear security awareness training (awareness meetings for senior officials) will be held regularly, IAEA is ready to support based on requests from Indonesia.

Based on IPPAS Mission results and follow up from IPPAS Mission report above, it can be said that Indonesia has implemented PPS at existing nuclear installation well. However, it is realized that the challenges in implementing PPS at Commercial NPP will be greater compared to research reactor because if a nuclear security issue happens at NPP, then it will impact the economy, health, psychology, politics, security and security aspects as well as Indonesian Government reputation in the world. One of the important things in this nuclear security issue is the importance of human resource capability, especially physical protection officer capability that is able to operate 24/7 at the commercial NPP. For PPS is operated 24/7, reliable personnel are needed (well trained) and equipped with training programs to improve skills and knowledges for the PPS personnel at the NPP. These trainings are highly needed by the physical protection officers, especially respond team, to secure NPP from unauthorized parties.

Physical protection system is an integration from human, procedure and equipment for the protection of assets or facilities against theft, sabotage, or other malevolent human attacks. BATAN has developed laboratory to train and for ~~tools~~ the introduction on physical protection system elements for detection, delay and response. In this laboratory, research and development are implemented on physical protection system elements. With the building commercial-scaled nuclear power plant plan from the government, there is an opportunity in the industrial field, such as a production of security equipment for the physical protection system for nuclear power plants. Industries can participate in effort to produce and develop physical protection elements so Indonesia can meet the physical protection system by itself.

Commercial-Scaled NPP Management has to meet the physical protection system requirements according to international recommendation and regulatory body regulation. In some of the nuclear industries in the world, physical protection officers at NPP generally come from security companies that are specifically providing security services for nuclear installation. The implemented criteria for those physical protection officers meet the regulatory body provision and regulation, particularly safety and security issues. The trust that are given to the security officers must be assured because NPP are vital objects. Trustworthiness implementation or Human Reliability Programee should be strictly implemented by nuclear security officers, and other employees at the facilities. This program is determined in CPPNM and INFCIRC/225/rev.5/2011.

NPP need a thorough consideration on safety, security and safeguard. Safety, of course, aims to prevent accident. And security aims to prevent intentional and dangerous incident to NPP or causing nuclear material theft and sabotage. Also, safeguard is an effort to prevent nuclear material deviation for nuclear weapon or bomb. Even with different safety and security activities, they have different focuses, they interact and collide each other. The taken action on one activity will cause implication on the other activities. The radioactive material releases to environment has been the main concern. With that, a special emphasis on NPP operation safety and security is needed. After 9/11 terrorist attacked and the latest terrorist activities that has happened all over the world, the operators body (Permit Holders), regulators and international organizations have enhanced the concern on ensuring an adequate protection at critical points of the facility. In that matter, NPP has been the special focus on this effort, because it is realized that any potential terrorist attack at NPP could cause damage in the community, as well as environmental contamination.

According to IAEA-International Nuclear Safety Group (INSAG), sophisticated and comprehensive nuclear safety regime development that has been done since years ago has the benefit in operating NPP. Even though now nuclear security issues are increasingly concerning, nuclear security regime development for NPP is still slow compared to nuclear safety regime. Seeing the nuclear security regulation revision progress and improvement that coincided with the demand from the public, the same standard on nuclear safety and security is needed. The challenges to meet those demands are predicted grow along with the interest on establishing the new NPP, both for the countries that has run NPP and for the countries that has planned to embarking of nuclear power plants.

IAEA has issued an Interface Between Safety and Security (INSAG-24) Year 2010 guide. INSAG put efforts on exploring challenges, especially those that are related to nuclear security emphasis elaboration, with focuses on safety and security connections at NPP. Nuclear safety and nuclear security have common goals to protect of human, community and environment from the impact of radiation hazards. **FIGURE 1** (Source: CITS-UGA) shows interaction between nuclear safety and nuclear security.

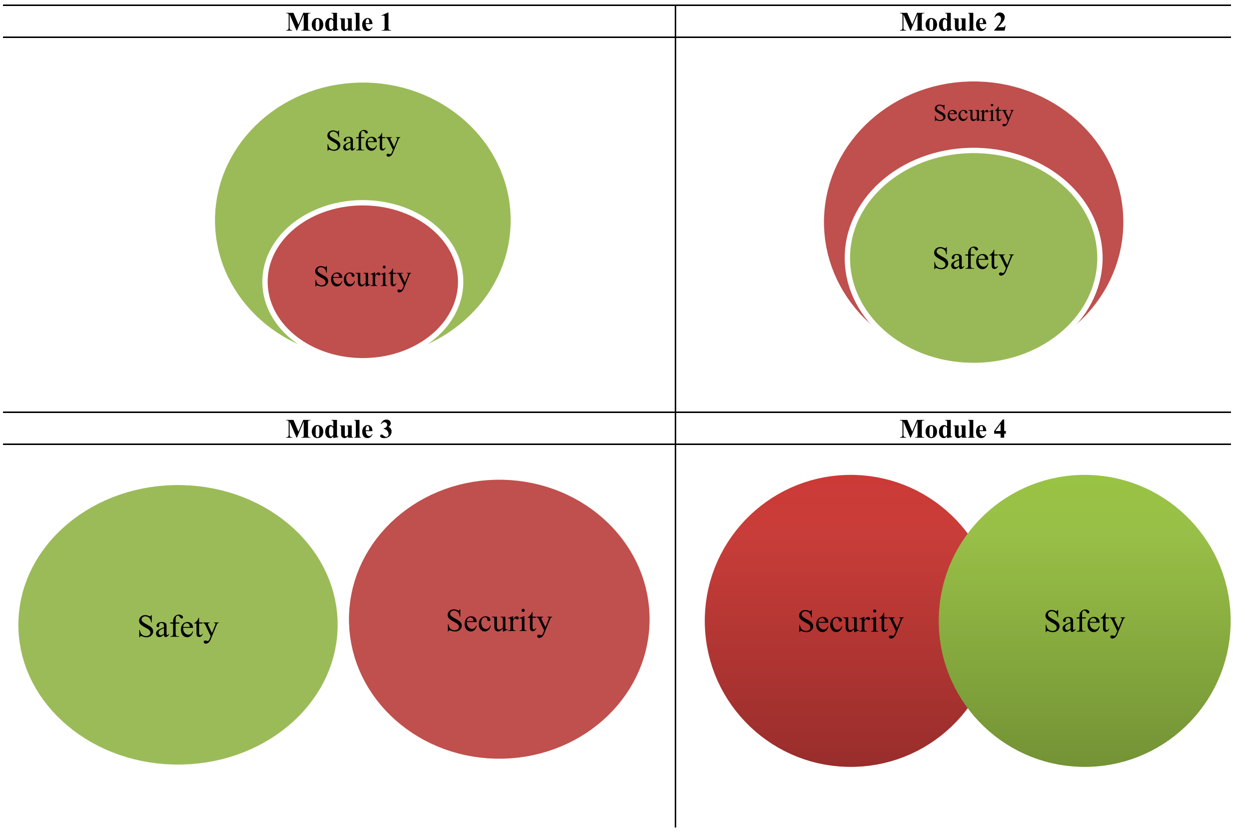


FIGURE 1. Safety and Security Interaction

Security culture has begun to be implemented in a structured way in Indonesia since 2010. Nuclear security culture internalization on all nuclear installation management levels, especially in BATAN refers to IAEA Nuclear Security Series Number 7, 2008 on Nuclear Security Culture (Implementing Guide). Regarding that matter, to study nuclear security implementation effectiveness and nuclear security culture at nuclear installation, Indonesia has performed self-assessment on nuclear security culture (first trail in 2012 and second self-assessment in 2015). In this case, Indonesia is one of the pioneer countries that perform self-assessment on nuclear security culture. That self-assessment performed according to the IAEA recommended guide. Until now, self-assessment has been performed three times in some Nuclear areas and installations managed by BATAN. The result of that self-assessment shows that generally nuclear security culture has been implemented well (with score of 5.25 of 7), even though some security culture characteristics still need to be improved (need references from ICONS 2013 paper and IJNS 2017 journal). In this security culture frame, including performing self-assessment, BATAN established Center for Security Culture and Assessment (CSCA) in 2014. With the CSCA, nuclear and radioactive material security culture dissemination and internalization as well as security culture implementation in Indonesia has gone through variety of trainings and well implemented. This matter will be the contribution for a better nuclear security infrastructure preparation when Indonesia establish NPP later on.

CONCLUSION

Global nuclear security regime is a deal made by IAEA member countries that agree on Convention on the Physical Protection of Nuclear Material changes (CPPNM, Amendment 2005). Indonesia is committed to implement global nuclear security regime, especially physical protection system implementation at research reactor facilities. Indonesia is whole-heartedly implementing global nuclear security regime that is reflected in some of the binding ratifications on regulations into government regulations. Also, IAEA physical protection evaluation team is welcomed by Indonesia in IPPAS mission to assess the physical protection implementation compatibility. These three IPPAS assessments show that Indonesia is able to manage PPS at the nuclear installations well. On the other hand, nuclear security culture self-assessment also shows that nuclear security culture improvement, especially at the nuclear installations, shows a good result.

Even so, there are some aspects that need to be prepared in accordance with commercial-scaled NPP development, which are Human Resources preparation that are ready for safety system at NPP. This Human Resources preparation can be implemented when the Indonesian government agree on the NPP development. In this context, based on the existing facilities and capabilities, BATAN is ready to provide education and trainings. Also, awareness and synergy for all nuclear security stakeholders need to be reinforced because commercial-scaled NPP is a national vital facility and all stakeholders need to be involved.

BIBLIOGRAPHY

[1] M. Muchlis and A. D. Permana, "Proyeksi Kebutuhan Listrik PLN Tahun 2003 s.d. 2020," *Pengembangan Sistem Kelistrikan dalam Menunjang Pembangunan Nasional Jangka Panjang,* pp. 19-29, 2003.

[2] International Atomic Energy Agency, Evaluation of the Status of National Nuclear Infrastructure Development No. NG-T-3.2 (Rev. 1), Vienna: International Atomic Energy Agency, 2016.

[3] International Atomic Energy Agency, IAEA Nuclear Security Series No. 10 - Development, Use and Maintenance of The Design Basis Threat, Vienna: International Atomic Energy Agency, 2009.

[4] International Atomic Energy Agency, "INFCIRC/274/Rev.1/Mod.1 - Amendment to the Convention on the Physical Protection of Nuclear Material," International Atomic Energy Agency, Vienna, 2016.

[5] United Nations, "INTERNATIONAL CONVENTION FOR THE SUPPRESSION OF ACTS OF NUCLEAR TERRORISM," United Nations, New York City, 2005.

[6] United Nations, "Security Council Resolution 1373," United Nations, New York, 2001.

[7] United Nations, "Security Council Resolution 1540," United Nations, 2004.

[8] International Atomic Energy Agency, "CODE OF CONDUCT ON THE SAFETY AND SECURITY OF RADIOACTIVE SOURCES," International Atomic Energy Agency, Vienna, 2004.

[9] "Peraturan Presiden Republik Indonesia Nomor 46 Tahun 2009 Pengesahan Amendment to The Convention on The Physical Protection of Nuclear Material (Perubahan Konvensi Proteksi Fisik Bahan Nuklir)".

[10] International Atomic Energy Agency, IAEA Nuclear Security Series No. 20 - Objective And Essential Elements of A State's Nuclear Security Regime Nuclear Security Fundamentals, Vienna: International Atomic Energy Agency, 2013.

[11] Badan Pengawas Tenaga Nuklir, "Peraturan Kepala Badan Pengawas Tenaga Nuklir Nomor 1 Tahun 2009 tentang Ketentuan Sistem Proteksi Fisik Instalasi dan Bahan Nuklir," Badan Pengawas Tenaga Nuklir, Jakarta, 2009.

[12] M. L. Garcia, Design and Evaluation of Physical Protection Systems: Second Edition, Oxford: Butterworth-Heinemann, 2007.

[13] International Atomic Energy Agency, IAEA Nuclear Security Series No. 7 - Implementing Guide - Nuclear Security Culture, Vienna: International Atomic Energy Agency, 2008.

[14] F. Hermana, K. Khairul, I. Khripunov and D. Nikonov, "Nuclear Security Culture in Practice," in *International Conference on Nuclear Security: Sustaining and Strengthening Efforts*, Vienna, 2013.

[15] "Peraturan Pemerintah Republik Indonesia Nomor 54 Tahun 2012 tentang Keselamatan dan Keamanan Instalasi Nuklir".
[16] "Peraturan Pemerintah Republik Indonesia Nomor 43 Tahun 2006 tentang Perizinan Reaktor Nuklir".
[17] "Peraturan Pemerintah Republik Indonesia Nomor 33 Tahun 2007 tentang Keselamatan Radiasi Pengion dan Keamanan Sumber Radioaktif".
[18] International Atomic Energy Agency, IAEA-Nuclear Security Series No. 13 - Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Rev.5), Vienna: International Atomic Energy Agency, 2011.
[19] A. R. Antariksawan, K. Khairul, H. Umbara and E. Kristuti, "Conducting Nuclear Security Culture Self Assessments in Nuclear Research Facilities Using the IAEA Methodology," in *Badan Tenaga Nuklir Nasional*, Jakarta , 2018.
[20] International Atomic Energy Agency, INTERNATIONAL PHYSICAL PROTECTION ADVISORY SERVICE (IPPAS) GUIDELINES, Vienna: International Atomic Energy Agency, 2014.
[21] IPPAS Guidelines-IAEA Services Series 29, Vienna, November 2014

Radiation Safety Assessment on The Use of Portable X-Ray for General Radiography. Case Study: Portable X-Ray of X-Manufacture

Titik Kartika[1, a)], Leily Savitri[2]

[1]Nuclear Energy Regulatory Agency of Indonesia (BAPETEN), DKI Jakarta 10120, Indonesia

[a)] Corresponding author: t.kartika@bapeten.go.id

Abstract. The new development science and technology of diagnostic radiology for general radiography using portable X-ray equipment shows that the need for precise diagnostics, easy and affordable services, is increasingly taken into account. Generally, portable X-ray equipment is not considered for general radiology examinations because the resulting image is not adequate, reducing medical practitioners' accuracy in conducting medical diagnoses. It can happen because portable X-ray equipment generally has lower energy than fixed X-ray equipment. The low power of portable X-ray equipment has portable X-ray equipment limited only to specific organs. But, recently, because of the advances in science and technology, there are various types of portable X-ray equipment in the field that are much better than before and meet radiation protection and safety aspects. It can assist the needs of medical services that are fast, practical, and efficient. A comprehensive study has been carried out regarding the use of portable X-ray equipment for general radiography regarding radiation protection and safety aspects. This study shows that portable X-rays are adequate to be used in general radiography if considering radiation protection and safety, especially considering its risks.

Keywords: portable X-ray equipment, radiation protection and safety, science and technology, efficient, risk

INTRODUCTION

X-ray equipment technology in the diagnostic radiology field continues to be developed to meet the needs of good, easy, and efficient patient diagnostics services. One form of technological development is the emergence of portable X-ray equipment with a small size but can produce better quality images. Following the characteristics that are easy to carry and move, portable X-ray equipment is designed to meet the needs of patients who cannot come or move to the radiology room. However, besides its benefits, some risks must be considered from the use of portable X-ray equipment. Thus, the development of portable X-ray equipment technology must be accompanied by radiation protection actions to ensure radiation safety for patients, workers, and public members.

Nowadays, there is a kind of portable X-ray equipment that its shape and size are used for general radiography examination. One of them is portable X-ray equipment of X-Manufacture. The X-ray equipment can be operated by holding the X-ray equipment directly with hands without a tube stand. It can reduce the optimization of radiation protection safety because there is a possibility of vibration or instability when the X-ray equipment is operated. As a result, it will reduce the quality of the resulted image. Moreover, there is a possibility that the operator's hands will get unnecessary exposure. Besides, several other things should also be considered related to the risk of using this portable X-ray equipment.

The purpose of this paper is to provide a comprehensive study of the use of portable X-ray equipment of X-Manufacture for general radiography in terms of radiation protection and safety aspects.

Objective

The objective of this assessment is to provide a comprehensive study of the use of portable X-ray equipment of X-Manufacture for general radiography in terms of radiation protection and safety aspects.

MATERIAL & METHOD

This study's methodology was carried out through literature studies from national and international references and reviewed secondary data from one of the portable X-ray equipment manufacturers (X-Manufacture).

Some data used in this study is the data from portable X-ray equipment of X-manufacturer, which will be used for general radiography. The portable X-ray equipment is digital radiographs in the form of a camera equipped with a detector and notebook 17 x 17 cm, mini cradles with varying angles. The X-ray equipment tube is fitted with a collimator with one field illumination size (collimation size cannot be adjusted) with the following technical specifications.

TABLE 1. Technical Specifications of X-Ray Portable Equipment of X-Manufacture

Parameter	Value
Focal spot size	0.4 mm
kV/mA of Tube	40 – 60 kV (adjustable) / 2 mA
output energy	120 W
energy supply	DC 11.1 V (Battery)
input energy	DC 5 – 12 V / 2.1 A (X-Ray unit)
	AC 100 – 240 V, 50 – 60 Hz / 1 A
Weight	1.8 Kg

FIGURE 1 below explains the results of radiation exposure measurements around the portable X-ray equipment of X-manufacture in 0.5 seconds at a distance of 10 cm, 20 cm, and 30 cm from the equipment's surface.

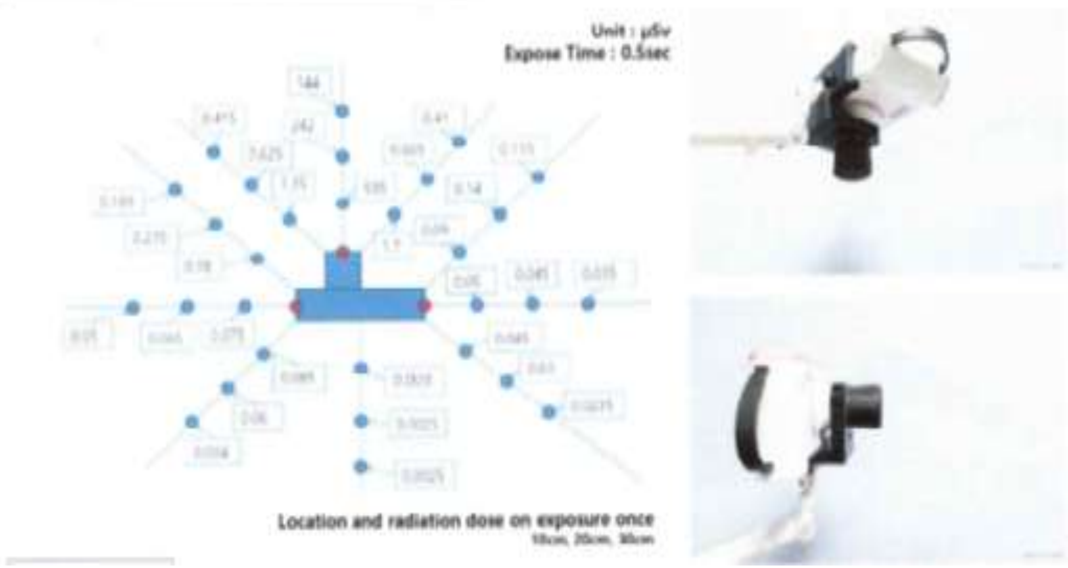


FIGURE 1. The radiation exposure around the portable X-ray equipment of X-manufacture [1]

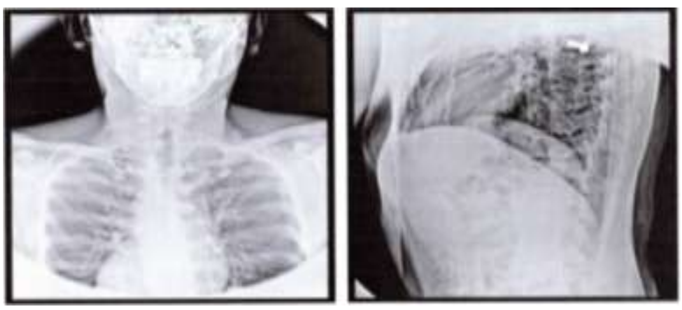


FIGURE 2. Chest image resulted from the portable X-ray equipment of X-manufacture [1]

FIGURE 2. shows the chest image resulted from the portable X-ray equipment of X-manufacture. Figure 3 shows the image's comparison resulting from X-ray portable of X-manufacture with the image resulted from general X-ray radiography.

The diagnostic effectiveness of portable X-rays for patients in the intensive care unit was reported to be 84.5%. Meanwhile, patients' radiological assessment in nursing homes is sufficient, with good image quality, and favorable factors such as patient safety and comfort, no need for transportation, and no need for staff to be absent.

One in 123 patients (241 radiographs) had to undergo repeat radiographs in the hospital because the images were unclear while the rest were sufficient for diagnosis. Cellular digital chest X-ray is sensitive and specific in detecting pulmonary tuberculosis in culture-confirmed cases. For comparison, the cellular X-ray has a sensitivity of 81.8% (95% confidence interval 64.5 to 93.0) and specificity. 99.2% (95% CI 99.1 to 99.3). [2]

FIGURE 3. The image resulted from X-ray portable of X-manufacture (left), and the image resulted from general X-ray radiography (right) [1]

RESULTS AND DISCUSSION

From the calculation of radiation exposure measurement that showed in Figure 1, the dose rate is at a distance of 10 cm from the focal spot surface with a time of 0.5 seconds in the position with the smallest dose (0.003 μSv) is 21.6 μSv/hour. It exceeds the limit that is required in Government Regulation No. 29 of 2008 (1 μSv/hour). Government Regulation No. 29 of 2008 Article 72 contains provisions concerning ionizing radiation generator which are exempted from the license. It states that under normal operating conditions, the equivalent dose rate in all directions does not exceed one μSv / hour at 10 cm from the equipment's surface. The maximum energy produced is less than or equal to Five keV [3].

Based on Government Regulation No. 29 of 2008 Article 72, the portable X-ray equipment of X-manufacture, is not included in the use of exempted from the license. The equivalent dose rate in all directions exceeds one μSv/hour at 10 cm from the equipment's surface. Therefore the portable X-ray equipment of X-manufacture must have a license from BAPETEN.

In terms of its physical characteristics, mobile X-ray and portable X-ray equipment are easy to move. They are designed to meet the needs of examinations of patients who cannot come or move to the radiology room: patients in the emergency unit room, patients in the intensive care unit room, patients with disabilities or patients who are difficult to move, patients in prisons, or patients in military operations [2, 4,5].

It is necessary to make provisions or guidelines for optimizing radiation protection and safety to use portable X-ray equipment to ensure radiation protection and safety. Besides, the examinations using portable X-ray equipment cannot be carried out without medical practitioners' proper justification [4,5].

Portable X-ray equipment is often moved to anywhere, so it is essential to have provisions on tracking the locations of portable X-rays equipment to supervise the safety of its uses. It can be executed by restricting the location area following those stated in its license. This area may include districts, cities, or provinces. Besides that, there should be a reporting system periodically to BAPETEN on the location of portable X-ray equipment. However, supervision related to the area should be coordinated with the Ministry of Health or the local Agency of Health because portable X-ray equipment is one form of health service that must be integrated with health facilities.

In addition to area restrictions, provisions need to be made to prevent the possibility of portable X-ray equipment used by unauthorized persons. The requirements regarding the place and the method to store portable X-ray equipment appropriate and safe, and the responsible person for controlling the storage and use of portable X-ray equipment are needed. It also should be clarified and considered who justifies using portable X-ray equipment and who analyzes the image. As radiologists should interpret the image over long distances, it is necessary to ensure adequate computer and internet network specifications. The image can be read and analyzed quickly and accurately, and will not inhibit the communication between the personnel examining with the radiologists. To ensure that portable X-ray equipment is operated by qualified personnel, proof of competency or expertise of each personnel is necessary.

The things that should be considered in portable X-rays related to the radiation protection and safety aspects that include the principles of justification, dose limitation, and optimization of radiation protection and safety are as follows.

Justification

Government Regulation No. 33 of 2007 Article 22 states that justification must be based on the benefits that are greater than the risks [6]. The advantages of using portable X-rays are providing health services to patients more efficiently, practically, and quickly, especially for patients who can not move to a fixed X-ray equipment room. For example, patients in emergency units, intensive care units, patients with disabilities, or critical patients who have difficulty or are significantly at risk when moving, patients in prison, or patients in military operations [3,4]. However, despite its benefits, its risks should be considered in the use of portable X-ray equipment. These risks that should be regarded as include [2, 4, 5]:

- The use of portable X-ray equipment can be done in an outdoor area, or in a room that does not have adequate shielding, or in a place where there are other patients or the public who are near portable X-ray equipment during exposure.
- Portable X-ray equipment is often moved, assembled, and stored anywhere, so it may be affected by mechanical stability that may disturb the generator's output.
- Portable X-ray equipment is small and easy to carry to be used or operated by unauthorized persons.
- Portable X-ray equipment uses batteries more often, so there is limited energy availability, which will affect the exposure process and image quality.
- The limited current (mA) used in portable X-rays can affect image quality. To get the expected image quality, it takes more prolonged exposure, and the radiation will be more significant.
- The limited kV used in portable X-rays also limits the examination types because some examinations require quite large kV parameters.
- The use of X-ray equipment meets the needs of patients who can not come to the hospital, so it takes time, place, and personnel to go to the site. It is necessary to consider the issue of resources related to personnel, funding for transportation, personnel, and quality control of the equipment that may be greater from the routine examination.
- In addition to the justification by BAPETEN, it should be considered the suggestions from radiologist professional organizations and the Ministry of Health. It relates to the readiness and improvement of human resource competencies, increased legal awareness, and justification for new technologies and procedural techniques. Request for consideration of justification of new technology to the protection organization and the ministry of health is a mandate of the General Safety Requirements Part 3 (GSR Part 3) of IAEA [7].

Because of its risks, the portable x-ray equipment should only be used for examinations where it is impractical or not medically acceptable to transfer patients to a fixed unit. The medical practitioners should justify the use.

Dose Limit

The radiation exposure around the portable X-ray equipment of X Manufacture, when operated, should be considered to estimate whether the dose value received by radiation workers exceeds the dose limit for radiation workers determined by the Bapeten Chairman Regulation No.4 of 2013 or not. The dose limit for workers is the effective dose of 20 mSv (twenty millisieverts) per year, on average, for 5 (five) years in a row [8].

From **FIGURE 1**, we get the dose calculation received by the radiation worker for a year. The dose is the highest dose behind or next to the X-ray equipment, where the operator is likely to stand, dose value is 0.085 μSv at a distance of 10 cm for a single exposure. If it is assumed that one worker operates the portable X-ray equipment 20 times per day, then the dose to be received by the worker for one year (250 workdays) is:

$$0.085\ \mu\text{Sv} \times 20 \times 250 = 425\ \mu\text{Sv} = 0.425\ \text{mSv per year}$$

From the calculation, it is obtained the most massive dose at a position 10 cm behind the equipment by assuming there are 20 operations in a day, which is 0.425 mSv per year, which is still much lower than the dose limit for radiation workers (20 mSv / year). However, it is highly dependent on information about how many operations are usually done by workers. Therefore, the information regarding the workload analysis of radiation workers is needed. It is advisable to verify the measurement of radiation exposure during radiation using portable X-ray equipment using a phantom so that the radiation exposure measurement results also take into account the patient's scattering radiation.

Besides, another thing that must be considered is work area restriction, namely the control area and supervision area. Access to work areas where radiation is being used should be controlled to ensure doses to visitors are below the dose limits for the public. In a diagnostic radiology facility, the control area is the locations where the X-ray equipment is operated. Therefore, where portable X-ray equipment is placed can also be categorized as controlled areas during radiological procedures are being carried out. The site should be shielded and should be restricted, and there should be radiation warning signs indicate that X-ray equipment is being operated [9]. Following BAPETEN Chairman Regulation No. 4 of 2013, personnel in the control area should use individual dose monitor and radiation protective equipment [8]. The supervised area may involve areas surrounding the control area. The supervised site is not primarily based on the radiation exposure level, which in radiology diagnostic can be kept very low, but instead as a 'buffer zone' due to other individuals' potential to enter the X-ray area inadvertently and be exposed. Thus, this supervision area should also be marked [3].

Optimization of Radiation Protection and Safety

The principle of optimization of radiation protection and safety, as explained in Government Regulation No.33 of 2007, is The optimization of radiation protection and safety is an effort to achieve radiation exposures are as low as reasonably achievable, with economic, societal, and environmental factors taken into account [6]. Optimization also is a prospective and iterative process that requires qualitative and quantitative information. It means that the level of optimization would be the best possible under the prevailing circumstances. To achieve optimization of radiation protection and safety, among other things, the appropriate features of X-ray equipment and radiation protection and safety procedures. [9]

Portable X-Ray Equipment Features

Portable X-ray equipment should have the following features [9,10, 11,12]:

- high-frequency microprocessor generator systems
- Operating parameters for radiation generators that are clearly and accurately shown
- X-ray tubes with adequate filtration
- equipment that indicates clearly (visually and audibly) when the beam is on
- adjustable beam collimating equipment
- battery energy indicators
- adequate internet network
- tube stand which is relatively stable to vibrations
- means to detect immediately any malfunction of a single component of the system
- means to minimize the likelihood of unintended or unnecessary exposures
- X-ray equipment radiation leakage does not exceed one mGy (one milligray) per hour at 1 (one) meter from focus.

Following the characteristics of portable X-ray equipment that are often installed, stored, and carried, there will be a possibility of changes in mechanical stability that are likely to affect the output's stability. Therefore, the portable X-ray equipment's internal quality control should be carried out more frequently than fixed X-rays equipment. External quality control should be done to ensure the tube's compatibility with the generator output and to ensure the reliability of the X-ray equipment. External quality control should be done through the compliance test mechanism based on the regulation on the compliance test of X-ray equipment interventional and radiology diagnostic.

Radiation Protection and Safety Procedure

Proper procedures will increase the optimization of radiation protection and safety. These procedures should be specified in the radiation protection and safety program document. The following are the radiation and protection safety procedures that should be carried out in portable X-ray equipment use [9].

- before portable x-ray equipment is used, it is necessary to determine and give boundaries the controlled and supervised areas to ensure that there are no unauthorized persons to enter the site around portable x-ray equipment. these boundaries should be marked.
- the operators should wear lead aprons.
- the operators' position should be behind the tube of portable x-ray. they should maintain as much distance as possible between themselves and the patient while still maintaining adequate visual supervision of the patient and communicating verbally with the patient (approximately at a distance of two meters).
- verbal warning of an imminent exposure should be given.
- in an area where other patients are adjacent to the examined patient, such in the emergency unit, mobile shields should be used. the primary beam should be directed away from staff and other patients whenever possible.
- other staff should be as far away from the patient as possible during the exposure (typically at least three meters) or are behind appropriate barriers.
- with a combination of distance, placement of mobile shielding, and careful control of the x-ray beam direction should ensure that appropriate public radiation protection is being afforded.
- for patient safety, it should be considered the diagnostic reference level, keep the distance between the x-ray tube and patient at least 1 m (one meter), and the collimator should be adjusted to patient examination needs.
- the operator's workload should be considered such that the dose limit is not exceeded, and the optimization of radiation protection and safety can be achieved.

CONCLUSION

This study provides some conclusions that the development of science and technology makes portable X-rays equipment adequate to be used in general radiography. The portable X-ray equipment should only be used for examinations where it is impractical or not medically acceptable to transfer patients to a fixed unit. Medical practitioners should justify their use. The use of portable X-ray equipment should consider the aspects of radiation protection and safety that cover the justification, dose limitation, and optimization principle. The risks that should be considered in using portable X-rays are the inadequate shielding of the working area, the mechanical instability, the limited availability of energy, the possible misuse by unauthorized or incompetent persons, and difficulties for the Regulatory Body in monitoring.

REFERENCES

[1] Seoul National University, Bundang Hospital, 2018, Portable X-Ray Mine.
[2] Canadian Agency for Drugs and Technologies in Health, 2016, Portable versus Fixed X-ray Equipment: A Review of the Clinical Effectiveness, Cost-effectiveness, and Guidelines, Rapid Response Report, Summary with Critical Appraisal, Kanada.
[3] Government of the Republic of Indonesia, 2008, Government Regulation No. 29 of 2008 on Licensing of the Utilization of Ionizing Radiation and Nuclear Materials (State Gazette of the Republic of Indonesia of 2008 Number 54, Supplement State Gazette of the Republic of Indonesia Number 4839).
[4] European Academy of Dento Maxillo Facial Radiology (EADMFR), 2015, Justification, and good practice in using handheld portable dental X-ray equipment.
[5] H. Umar, S. Shaffaque; Lahore, Punjab, 2015, Roles, and Responsibilities of Medical Imaging Technologist in Justification of Portable X-Rays Examinations, ECR.
[6] Government of the Republic of Indonesia, 2007, Government Regulation Number 33 of 2007 onThe Safety of Ionizing Radiation and The Security of Radioactive Source (State Gazette of the Republic of Indonesia of 2007 Number 74, Supplement State Gazette of the Republic of Indonesia Number 4730).
[7] International Atomic Energy Agency (IAEA), 2014, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, General Safety Requirements No. GSR Part 3, Vienna
[8] Nuclear Energy Regulatory Agency (BAPETEN), 2014, BAPETEN Chairman Regulation No. 4 of 2013 on Radiation Protection and Safety in Utilization of Nuclear Energy.
[9] International Atomic Energy Agency (IAEA), 2014, Radiation Protection and Safety in Medical Uses of Ionizing Radiation, Specific Safety Guide No. SSG-46, Vienna.
[10] Department of Health and Human Services, 2011, Office of Inspector General, Questionable Billing Patterns of Portable X-Ray Suppliers.
[11] AAPM, 2002, Quality Control in Diagnostic Radiology, Report No. 74.
[12] National Centre for Antarctic & Ocean Research, Tender Document for Supply of Medical Equipment Portable Digital 300 mA X-Ray Machine

Strengthening the Radiation Safety of Patients by Controlling and Preventing Unnecessary Exposure in Radiological Diagnostic and Interventional Facilities

Endang Kunarsih[1, a)]

[1] *Badan Pengawas Tenaga Nuklir, Jl. Gajah Mada No. 8 Jakarta Pusat*

[a)] e.kunarsih@bapeten.go.id

Abstract. Unnecessary exposure is the situation of radiation exposure that should not be received by the patient, either in whole or in part, when undergoing radiation medical treatment. This situation occurs frequently but the impact on the patient is not always observable directly, so it was usually unnoticed and even ignored. An analytical descriptive study has been carried out regarding unnecessary exposure control profiles in diagnostic and interventional radiology facilities to propose recommendations for improving patient radiation safety through controlling and preventing unnecessary exposure. The profile mapping result shows that unnecessary exposure control and prevention in RDI facilities generally have been carried out but it has not been systematically built and only a few facilities have fully implemented it. This situation is caused by the absence of a direct effect observable on the patient (such as injury) resulting in the neglect of all potential risks that may arise including the follow-up system to control and prevent it. It needs a clear regulatory framework and guidelines to encourage the proper implementation of unnecessary exposure control and prevention.

Keywords: unnecessary exposure, patient safety, medical exposure

INTRODUCTION

In medical exposure, the patient is part of the object of investigation or medical treatment using ionizing radiation sources for diagnosis or disease therapy. The radiation dose given to these patients cannot be limited by the dose limit value, but using other constraints, for example, the guidance level and justification. Therefore, the risk that a patient could potentially accept was unnecessary radiation exposure. These risks must be managed properly and even prevented.

In the context of patient safety, Law Number 44 Year 2009 concerning Hospitals, Article 43, states that hospitals are required to apply patient safety standards, including risk assessment, identification and risk management of patients, incident reporting and analysis, the ability to learn. and following up on incidents, and implementing solutions to reduce and minimize the occurrence of risk [1]. In the IAEA document, GSR Part 3 has provided requirements to minimize the possibility of incidents of unintended radiation exposure to patients and follow-up for incident prevention [2]. Concerning the risk of incidents arising from the use of ionizing radiation, Government Regulation (GR) Number 33 Year 2007 concerning Safety of Ionizing Radiation and Security of Radioactive Sources, Article 22 and Article 34, mandates the implementation of justification and optimization principle of radiation protection and safety in every utilization of nuclear power including in medical sector to reduce the risk of unnecessary medical exposure [3].

Unnecessary exposure to patients is a radiation exposure situation that should not be received by patients, either in whole or in part, when undergoing radiation medical treatment. This situation is one of the risks that will be faced if medical exposure is not justified and adequately optimized. For example, patients get radiation exposure in an undesirable position/location (for example in the exposure for the thorax, X-ray beam reaches to the head area), patients get repeated radiation exposure (for example because the image does not match what the doctor needs, the film image is unclear, the process of radiation interrupted due to power outages, or patient imaging data lost due to power outages or software interruptions), patients get high doses of radiation because personnel does not notice radiation/time exposure indicators, and individuals get thorax exposure for an annual medical check-up.

Unnecessary exposure incidents occur frequently but their impact on patients is not always observable directly because in general, it is a stochastic effect. This causes unnecessary exposure incidents were unnoticed and even ignored. However, if this situation occurs repeatedly it will be detrimental to patient safety and will reflect that the performance of the facility is inadequate quality. This situation needs to be improved immediately to ensure radiation protection and safety for patients.

A study was conducted regarding the profile of controlling and preventing unnecessary exposure in diagnostic and interventional radiology facilities in Indonesia and identifying recommendations that can be proposed to improve the control and prevention system for unnecessary exposure. This paper provides recommendations for health facilities to strengthen radiation protection and safety systems for patients and for regulators both BAPETEN and the Ministry of Health (MOH) in developing policies related to radiation protection and safety for patients.

METHODOLOGY

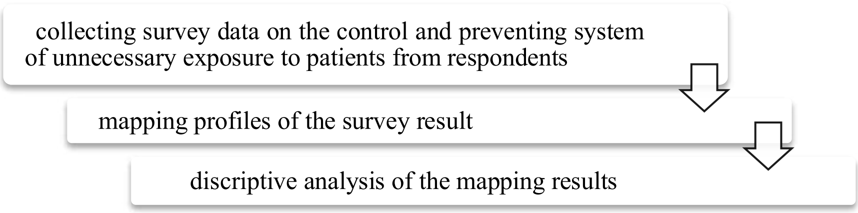


FIGURE 1. Methodology

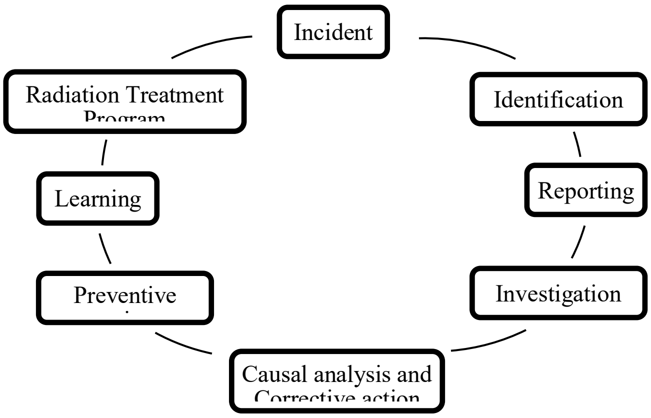


FIGURE 2. Incident learning systems [4]

As illustrated in **FIGURE 1**, the study began by collecting survey data on the control and preventing system of unnecessary exposure to patients from respondents. The respondent was 38 hospitals of radiology diagnostic and interventional facilities in the region of Central Java, Yogyakarta, Jakarta, Bandung, and Medan. Parameters in the questionnaire were compiled and summarized from some of the requirements in IAEA documents such as GSR part 3 and SSG 46 and regulations such as GR Number 33 Year 2007, BCR Number 4 Year 2013, and BCR Number 8 Year 2011. These parameters contained questions that describe the readiness of the system to control and prevent unnecessary exposure to patients for each facility. Then the mapping of the profile of the questionnaire results was carried out. Aspects of the mapping process and discussion follow the aspects of the incident learning system as presented in **FIGURE 2**.

RESULTS AND DISCUSSION

There were 38 questionnaires distributed to respondents, but only 27 questionnaires were filled out. The results of data processing are presented in **FIGURE 3, 4, & 5.**

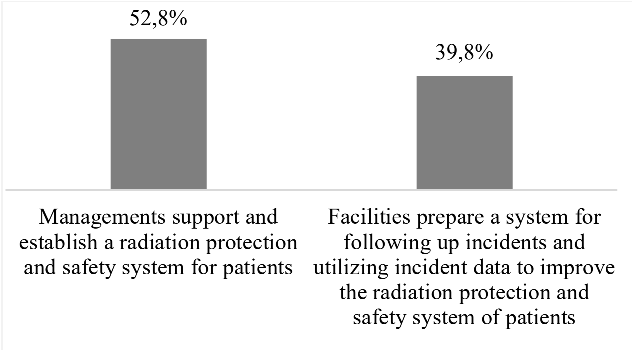


FIGURE 3. Profile of the unnecessary exposure control and prevention at the facilities.

Based on **FIGURE 3**, it appears that 52.8% of facilities have management supports for patient's radiation protection and safety and 40.3% of facilities have prepared a system for following up incidents and utilizing incident data to improve the radiation protection and safety system of patients. This means that controlling and preventing unnecessary exposure, in general, have been done but have not been applied systematically. Facility management generally supports, but only at the macro level. They only ensure the safety of hospital patients in general but do not cover the safety aspects of radiation.

Identification and Reporting

Unnecessary exposure incidents generally do not have any direct impact that can be observed, therefore the awareness of medical personnel to identify proactively is very important. But in **FIGURE 4** shows that less than 50% of facilities implement the identification system against the possibility of unnecessary exposure, i.e. 38.9% of facilities establish criteria of conditions that are categorized as unnecessary exposure to the patient, 44.4% of facilities are carried out the identification process routinely and 50.0% of facilities are documented the identification process through procedures and records. It reinforces the hypothesis that the absence of an observable impact directly ignores any potential risks that may arise.

Identification of the possibility of unnecessary exposure to patients should be done through monitoring and/or evaluation of the dose received by the patient and the examination process undertaken. This evaluation should be carried out by radiographers, medical physicists, and/or radiation specialists. Information or parameters related to patient doses can generally be obtained from visual displays on X-ray modality, from examination results (i.e radiographs image), and logbooks of irradiation conditions. The facility should locally set conditions criteria that can be categorized as unnecessary exposure so that they can be used as indicators in incident identification.

Information or parameters related to the patient's dose to be evaluated include [5], [6]: kerma area product (KAP), tube voltage, current irradiation time, length of time for exposure when termination of exposure fails, CT dose index average volume (CTDIvol), dose-length product (DLP), mean glandular dose (MGD), target/filter combination, breast compression thickness, cumulative air kerma both from the fluoroscopy process and from image acquisition, cumulative fluoroscopy time, number of fluoroscopy images recorded, and other relevant dose metrics. Data that are still in the form of radiation parameters should be converted into the patient's dose quantity by medical physicists and then be evaluated. Evaluation of patient doses is conducted by comparing patient doses to the available local or national Diagnostic Reference Level (DRL) values. If the comparison results show that the patient's dose is higher than DRL then unnecessary exposure may occur so that further analysis and corrective actions are needed.

Evaluation of the patient's radiograph also needs to recheck the suitability of the irradiation area and target image with the irradiation area and target requested by the referring physician. If there is any difference, for example, exceeds or less than or does not fit so as it is requested for re-exposure, the patient may get unnecessary exposure.

Identified incidents must be recorded and reported, as required in GSR Part 3 Requirements 9 paragraph 3.15 (g) that licensees shall establish procedures for reporting on and learning from accidents and other incidents [2]. However, based on Figure 4 shows that less than 50% of facilities that implement the recording and reporting of unnecessary exposure incidents, namely 38.9% of facilities that prepare a reporting scheme for unnecessary exposure to the patient and only 44.4% of facilities that documenting the reporting system into procedures and records.

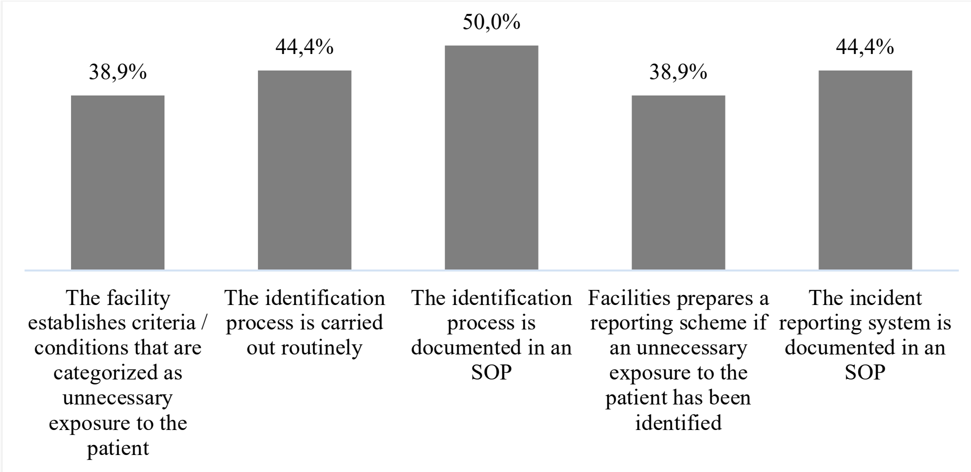


FIGURE 4. Profile of the identification and reporting implementation.

In general, the facility has prepared an incident reporting system as the mandate of the Decree of the Minister of Health of the Republic of Indonesia Number HK.02.02/MENKES/535/2016 concerning the National Committee for Hospital Patient Safety. In that regulation hospitals must prepare an incident reporting system that includes the establishment of policies, reporting flow, reporting forms, and reporting procedures [7]. However, the incident category used in that regulation was the patient with an injury. So that radiation incidents with unnecessary exposure types are not included in the reporting system because they are considered harmless. Also, some respondents think that reporting an incident is sometimes reluctant to do because they are afraid of getting sanctions or being opposed by other personnel.

Research from Hwang, et.al and Iskandar, et al have identified non-technical aspects that hinder reporting actions, which are as follows [8], [9]: blaming culture, legal sanctions or social sanction cause the incident to be deliberately covered up, lack of personnel's concern so reports are often late or reports are incomplete or even not reported, unclear reporting system so that the roles and responsibilities of the parties related to the reporting system are unclear, unclear conditions criteria to be reported so that they are not aware of situations that must be reported, high workloads so that they do not have time to do report and lack of management commitment in following up on reporting so that reporting does not have a positive influence on improving patient safety.

Therefore, efforts are needed to avoid those obstacles. One of them is conducting regular socialization of the reporting system to all hospital employees, especially those working in facilities that use ionizing radiation modalities. Besides that, training of personnel regarding the incident reporting system also needs to be done, which includes the purpose and benefits of the report, reporting flow, how to fill out the reporting form, when time to report, the notions used in the reporting system and how to analyze the report [8]. A reporting system needs to build, for example, an online-based electronic reporting system (e-reporting system) platform. BAPETEN can act as a promoter to build an online-based reporting system that facilitates the reporting of radiation incidents in radiological facilities, including unnecessary exposure to patients. In addition to ease of reporting, this system will also function as a platform for joint learning to optimize radiation protection and safety for patients in medical exposure situations.

Although unnecessary exposure incidents to patients have not led to emergency exposure situations or cause injury to patients, recording, and reporting are important as an effort to improve the system of patient's radiation safety. Valid and accurate incident data records will determine the evaluation accuracy of the patient's radiation safety system, underlie improvements in the service system based on patient radiation safety, and prevent the recurrence of radiation incidents in patients [10].

In the case of unnecessary exposure incidents, reporting should be addressed to the physician in charge, the head of the installation or management at the level above it, the team related to patient safety, or based on the reporting hierarchy set by the hospital. Information related to the incident unnecessary exposure to the patient, although it does not cause injury or other severity effects, must also be notified to the referring physician and the patient himself or the patient's family [2].

Investigation, Cause Analyze, and Corrective Actions

The unnecessary exposure incidents that have been identified, recorded, and reported to management should be investigated as required in GSR Part 3 Requirements 41 that licensees shall prompt any investigations such as incident exposures and, if appropriate, shall implement corrective actions [2]. The investigation process is designed to provide an explanation of the specific underlying cause of the incident and produce recommendations for following up and ensuring resolution for each root cause [4]. However, based on **FIGURE 5**, it appears that only 44.4% of facilities conducted investigation and evaluation of root causes, 27.8% of facilities documented investigation and evaluation of root causes process, and 38.9 % of facilities conducted corrective actions. In their opinion, unnecessary exposure is considered as a harmless incident or does not cause injury so no further follow up is needed.

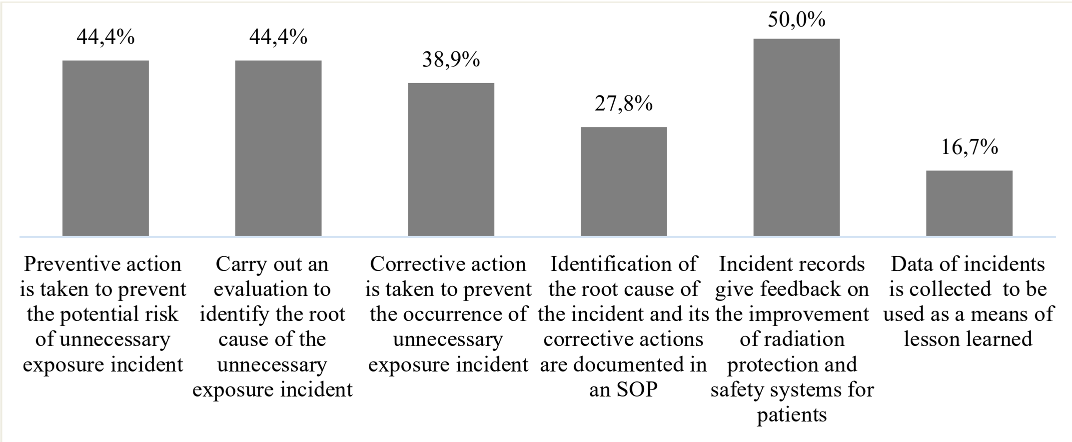


FIGURE 5. Profile of the incidents follow-up.

Although unnecessary exposure incidents to patients, in this context, do not lead to emergency exposure situations, it still needs to be investigated/analyzed using a structured approach such as Root Cause Analysis (RCA). A series of investigations and RCA should be carried out by the team so that the information collected and analysis point of view can be more comprehensive. Various tools can help to conduct RCA, such as fishbone diagrams and others. If the root causes of the problem have been identified, recommendations, plans, or strategies can be identified as the basis for implementing corrective actions to prevent the recurrence of the same incidents [11].

Based on survey data, it was identified that several causal factors that have contributed to unnecessary exposure are human and equipment/infrastructure as presented in **TABLE 1** [5], [12], [13]:

TABLE 1. The causal factors of unnecessary exposure incidents

Causal Factors	
Human	• Error in understanding prescriptions/requests from referring doctors.
	• Error in understanding examination protocols.
	• Error in understanding information displayed on the monitor or from the software
	• Error in identifying patients.
	• Error in recognizing dose indicator and error messages that appear on the monitor display (sometimes personnel even ignore it).
	• The patient uncooperative or moving when irradiated.
	• Error in setting irradiation target / location / position.
	• Use inconsistent quantities and units when measuring, testing, or calculating
Equipment and infrastructure	• Incorrect or outdated use of files, forms, protocols.
	• Digital image data is lost/erased before image evaluation is performed.
	• The radiograph/image is not clear.
	• Error in software upgrade affecting protocol and image processing settings.
	• The electrical power suddenly stopped (power outages).
	• Poor equipment reliability due to age and workload.
	• Malfunction of AEC (Automatic Exposure Control).
	• The AEC chamber is not in line with the X-ray tube.
	• Malfunction of the exposure timer.
	• Unavailability of a warning system related to an overdose.
	• The CT Scan position setting is reset so that the patient scanned in the wrong position
	• Internal parameters no longer match after the equipment is repaired
	• Technical errors in imaging systems, such as Picture Archiving and Communication Systems (PACS), and Radiology Information Systems (RIS)

Based on the causal factors identified in Table 1, it can be illustrated that the root causes of unnecessary exposure incidents are presented in **TABLE 2** [4], [5], [12], [13].

TABLE 2. The root causes of unnecessary exposure incidents

	Root Causes
Manage-ment	• Lack of commitment from management and radiation workers in implementing safety culture.
	• Communication problems, both vertical and horizontal communication, and communication to the patient.
	• Unclear functions and lines of authority and accountability.
	• Inadequate design assessment of ergonomic impacts and operational capabilities.
	• Inadequate resource requirements planning and risk assessment.
Personnel	• Inadequate training and education for personnel in terms of radiation protection and safety and technical or clinical topics related to their area of work including the operation/use of equipment.
	• Lack of review of the competence and availability of personnel after the purchase of new equipment and after the workload has increased.
	• Lack of supervision for inexperienced personnel.
	• Lack of personnel availability or high personnel turnover cycle.
	• Lack of awareness of workers for work responsibilities assigned (due to health conditions, motivation, fatigue, psychological pressure, etc.).
	• Doctors do not consider the signs and symptoms of patients or alternative medical measures that are more appropriate.
Protocol	• Inadequate protocols or operational procedures that are difficult to be understood by personnel so that they are not implemented or even violated.
	• The lack of operational protocols or procedures causes the wrong activities to be carried out.
Equipment	• Inadequate implementation of quality assurance and multi-layered defense systems, for example, periodic evaluations of protocols and quality control of equipment.
	• Lack of programs for acceptance testing, commissioning testing, and quality control of equipment (both treatment equipment and radiation protection equipment).

The root cause that has been identified will be the basis for establishing appropriate corrective action. Corrective action has the aim of eliminating the root causes of the nonconformity that have occurred so that the same nonconformity is not repeated [14]. However, in **FIGURE 5** it appears that only 38.9% of facilities followed up the investigation through corrective actions and documented the corrective action process. This is because unnecessary exposure is considered as a harmless incident or does not cause injury so it does not require to follow up. Some respondents even said that corrective action is a time-wasting activity that is only intended for accreditation activities. This means that there was no awareness that corrective action will be one of the efforts in improving the system of protection and safety of radiology patients.

Corrective actions can be formulated appropriately if the investigation or analysis of the causes is done properly as well. In the context of unnecessary exposure incidents, corrective actions that can be taken are presented as follows: [4], [5], [12], [13], [2]

- Quality control on each stage of the process routinely.
- Quality control of equipment, for example, X-ray modalities, supporting equipment (imaging systems, software, hardware, information systems, decision support systems), and radiation protection equipment.
- Establish clearly and detailed protocols/procedures for each process and activity including testing activities for quality control and activities for the justification process.
- Ensure the ability and updating of the protocol/procedure.
- Provide personnel training, with applicative technical topics under their area of assignment.
- Clearly define the roles, responsibilities, and functions of personnel in the radiology facility.
- Providing patient radiation dosimetry information, either through direct measurement, calculation, or from the dose indicator in the modality.
- Implementation of DRL.

Corrective actions plans and results of must be documented, usually combined with investigative or root cause analysis reports. This includes the parties assigned to execute corrective actions and the deadline. After implementing corrective actions, monitoring is needed to assess their effectiveness in eliminating repeated incidents of the same incident. If the same incident still happens this indicates that the investigation/analysis is not accurate enough to identify the right root cause or in planning the right corrective action.

Preventive Actions

Corrective action is a follow-up action if an incident has occurred. Facilities should not only focus on problems that have occurred but also need to pay attention to prevention systems. In the case of preventive measures, a facility must identify potential problems or identify risks that may occur during the implementation of activities to minimize or prevent potential problems or non-conformities. However, from Figure 5 it appears that only 44.4% of facilities carry out preventive measures. This is because unnecessary exposure is considered as a harmless incident or does not cause injury so it is not a priority in the risk assessment process, even some respondents did not include these parameters in the assessment of patient safety risk.

Prevention systems through the identification of potential problems or identification of risks must begin at an early stage. Based on the identification results, the facility can carry out mitigation actions that are integrated into the established quality management system. In general, steps that can be taken to establish preventive actions are as follows:

a) Identifying critical points in the process of radiation examination. Tools that can be used include a map of the process of the examination of patients in radiology facilities.
b) Identifying potential risks that occur at those critical points, including potential obstacles in achieving safety (safety barrier). For identifying risks appropriately, it is necessary to understand the factors that can influence the risk. In the case of unnecessary exposure to patients, factors that influence include human resources, equipment, materials, processes, working flow, work environment, and others. Tools that can be used for this activity are the 'fishbone diagram' method, the 'five whys' method, the '5W2H' method, the Health Care Failure Mode and Effect Analysis (HFMEA) method, and others.
c) Analyze the risks identified in step b). In this stage, all types of risks, sources of risk, root causes of risks, controls or existing protections, opportunities for the occurrence of risks, the consequences that may arise will be discussed in detail, and results will be reported as completely as possible.
d) Evaluate the level of risk based on the impact of the risk and the chance of the risk occurring.
e) Mapping and prioritizing risks that significantly affect the incidence of unnecessary exposure in patients.
f) Based on the analysis and assessment of risk from steps c) to e), then plans and actions are taken to anticipate and/or reduce and prevent risks to acceptable limits based on applicable regulations and standards.
g) Monitor the effectiveness of the implementation of preventive measures. In the context of unnecessary exposure to patients, effective criteria can be shown by decreasing the tendency of situations that can trigger unnecessary exposure to patients.

The series of a preventive process as described above should be carried out by the team so that the information/data collected can be comprehensive and viewpoints can be more diverse so that the solution will be easily accepted by all parties.

Based on discussions with respondents and studies from literature, actions to prevent potential incidents of unnecessary exposure based on the approach of justification and optimization principle of radiation protection are as follows [15] [13] [12]:

- Promote a safe and secure utilization of ionizing radiation modalities.
- Promote clinical decision-making processes based on complete information.
- Develop and foster the implementation of culture to work with awareness and alertness in the context of quality and safety.
- Provide clear and detailed protocols and procedures for each process and activity, including activities related to quality control.
- Provide educated and trained personnel at an appropriate level and in an adequate number.
- Conduct supervision for new personnel.
- Increase the professionalism of personnel through applicative technical training under their area of assignment.
- Clearly define the roles, responsibilities, authorities, and functions of each person in the radiation facility (doctors, medical physicists, radiographers, nurses, administrative staff, and others).
- Provide a complete quality assurance program.
- Provide adequate and reliable resources (personnel, equipment, infrastructure, software, and hardware, etc.) according to the needs identified through risk assessment.
- Enhancing patient education regarding safety culture in various ways, for example, socializing the importance of a patient history record card stored by the patient himself.
- Lesson learning of unnecessary exposure incidents that have occurred.

Lesson Learned

Lesson learned is considered as one of the most effective incident prevention efforts. Lesson learned is part of knowledge management, which is a knowledge artifact that states knowledge in the form of experience, applies to an activity, decision, or process that, if reused, will have a positive impact on organizational results [16]. However, Figure 5 shows that 50% of facilities realized that incident record data give feedback on the improvement of radiation protection and safety systems for patients but only 16.7% of facilities implemented lessons learned from the incident. This is because most DIR facilities do not have a system that facilitates lessons learned, which of course is due to the mindset that there is no risk in DIR facilities. To overcome this, regulators (BAPETEN and MOH) and professional associations must collaborate to promote a system of lessons learned from an incident, for example by building frameworks or providing support for the following efforts:

- Review of the effectiveness of the corrective actions taken and communicates lessons learned from the incident to personnel involved in the incident and the investigation phase, and to the wide audience, for example by staff meetings, management meetings, incident review meetings at all facilities, or meetings between similar organizations.
- Establishment of a team that has functions to regularly review the lessons learned from all incidents, at least once a year. The purpose of this review is to identify any improvements throughout the system including systems that might not be identified because the incident was previously investigated and considered separately. The results of this review will be communicated to all staff.
- Review of some lessons periodically that have been identified from past incidents, in the context of investigating more recent incidents.
- Establishment of an online-based incident lesson learned system as a learning platform to optimize radiation protection and safety for patients in medical exposure situations. This system is generally integrated with an online reporting system.

CONCLUSION

In the profile mapping, unnecessary exposure control and prevention in DIR facilities generally has been carried out but it has not been systematically built, which is shown from the aspect of implementing incident identification was 44.4%, incidents recording and reporting was 38.9%, investigation and analysis was 44.4%, corrective action was 38.9%, preventive action through risk control was 44.4% and lesson learned was 16.7%. This situation is caused by the absence of a direct observed effect on the patient (such as injury) resulting in the neglect of all potential risks that may arise including the follow-up system to control and prevent it.

Unnecessary exposure incidents can occur at any time repeatedly and have a detrimental effect on patients, therefore safety culture needs to be promoted in each stakeholder. Leaders or top management must arrange concrete steps in preventing and controlling unnecessary exposure to patients by using various effective approaches for each facility, for example, the 3A approach (awareness, appropriateness, audit).

BAPETEN, Ministry of Health, and professional association should support the efforts to control and prevent unnecessary exposure by establishing a clear regulatory and supervisory framework.

ACKNOWLEDGMENT

The study was financed by DIPA BAPETEN 2019 and carried out in cooperation with P2STPFRZR team for collecting data, respondents for joining the survey, and experts who provide technical advice.

REFERENCES

[1] *Undang Undang Nomor 44 Tahun 2009 tentang Rumah Sakit*. Indonesia, 2009.

[2] IAEA, *General Safety Requirements Part 3 No. GSR Part 3: Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards*. Vienna: IAEA, 2014.

[3] *Peraturan Pemerintah Nomor 33 Tahun 2007 tentang Keselamatan Radiasi Pengion dan Keamanan Sumber Radioaktif*. Indonesia, 2007.

[4] D. Cooke *et al.*, *A Reference Guide for Learning from Incidents in Radiation Treatment*, vol. 22. Canada: Alberta Heritage Foundation for Medical Research, 2006.

[5] C. J. Martin *et al.*, "Unintended and accidental medical radiation exposures in radiology: Guidelines on

investigation and prevention," *J. Radiol. Prot.*, vol. 37, no. 4, pp. 883–906, 2017.
[6] U.S. Environmental Protection Agency, *Federal Guidance Report No. 14-Radiation Protection Guidance for Diagnostic and Interventional X-Ray Procedures*. Washington DC, 2014.
[7] Kementerian Kesehatan, *Keputusan Menteri Kesehatan Nomor HK.02.02/MENKES/535/2016 tentang Komite Nasional Keselamatan Pasien Rumah Sakit*. Indonesia, 2016.
[8] KKP-RS, *Pedoman Pelaporan Insiden Keselamatan Pasien*. Jakarta: Kementerian Kesehatan, 2015.
[9] H. Iskandar, H. Maksum, and Nafisah, "Faktor Penyebab Penurunan Pelaporan Insiden Keselamatan Pasien Rumah Sakit," *J. Kedokt. Brawijaya*, vol. 28, no. 1, pp. 72–77, 2014.
[10] J. I. Hwang, S. IL Lee, and H. A. Park, "Barriers to the Operation of Patient Safety Incident Reporting Systems in Korean General Hospitals," *Healthc. Inform. Res.*, vol. 18, no. 4, pp. 279–286, 2012.
[11] A. W. Wu, A. K. M. Lipshutz, and P. J. Pronovost, "Effectiveness and Efficiency of Root Cause Analysis in Medicine," *JAMA - J. Am. Med. Assoc.*, vol. 299, no. 6, pp. 685–687, 2008.
[12] IAEA, *Specific Safety Guide No. SSG-46: Radiation Protection and Safety in Medical Uses of Ionizing Radiation*. Vienna: IAEA, 2018.
[13] I. Turai and K. Veress, "Radiation Accidents : Occurrence, Types, Consequences, Medical Management, and the Lessons to be Learned," *Cent. Eur. J. Occup. Environ. Med.*, vol. 7, no. 1, pp. 3–14, 2001.
[14] ISO, *ISO 9000:2015 Quality management systems - Fundamentals and vocabulary*, 4th ed. Switzerland: ISO, 2015.
[15] FDA, "Initiative to Reduce Unnecessary Radiation Exposure from Medical Imaging," 2010.
[16] M. N. A. Sharif, N. H. Zakaria, L. S. Ching, and L. S. Fung, "Facilitating Knowledge Sharing Through Lessons Learned System," *J. Knowl. Manag. Pract.*, 2005.

Measurement of Gamma and Neutron Ambient Dose on The Outer Wall of the 18 Mev Cyclotron Shielding

B.Y. E. B. Jumpeno[1*], R. Tursinah[2], Nazaroh[1], I. Taufiq[1], R. A. Tuloh[3]

[1]*Center for Technology of Radiation Safety and Metrology, National Nuclear Energy Agency, Jl.Lebak Bulus Raya No.49 Jakarta, Indonesia*
[2] *Center for Applied Nuclear Science and Technology, National Nuclear Energy Agency, Jl. Taman Sari No. 71 Kota Bandung, Indonesia*
[3] *Muchtar Ryadi Comprenensive Cancer Center Siloam Hospital Semanggi, Jl. Garnisun No. 2-3, Jakarta, Indonesia*

[1*]Corresponding author: jumpeno@batan.go.id

Abstract. In the last decade positron emission tomography (PET) has been widely used for imaging organ function. In a cyclotron, accelerated proton interaction with the target or the component material produces neutron and gamma radiation. The use of cyclotrons for medical purposes is necessary to monitor the exposure to neutron and gamma radiation. In this study, measurement of gamma dose, neutron dose and its spectrum were carried out on the outer wall of the 18 MeV cyclotron PET radiation shielding. Gamma dose measurements were performed using four environmental OSL dosimeters. The dosimeters are wrapped in aluminum foil to prevent the Al2O3:C element from being exposed to light and mounted on the outside of the entry bunker wall, power supply wall, and general store wall of cyclotron shielding 18 MeV IBA MRCCC, Siloam. Measurement of neutron is done using a BSS device mounted on the measurement position of the outer wall of the bunker entry and the outside wall of the general store. The highest dose of gamma obtained in the outer position of the shielding was (367 ± 10.6%) μSv/3 months. The result confirmed that the thickness of the 220 cm concrete shielding construction is capable of absorbing the generated gamma radiation. The result of neutron calculations show that the largest ambient dose rate comes from fast neutron in the outside position of the wall of the bunker entry was (0.475 ± 12%) μSv/h, while in the outside position of the general store wall was (0.331 ± 12%) μSv/h. The epithermal and thermal neutron of ambient dose rate, however were very small.

Keywords: ambient dose, gamma, neutron, shielding, cyclotron.

INTRODUCTION

The use of nuclear technology in the health sector has been increasing rapidly. During the last few decades, PET (*positron emission tomography*) has become a widely used functional imaging technique (Mendez *el.al. 2005).* A cyclotron is an accelerator machine that accelerates particles in a circle so that high kinetic energy is obtained. These particles can be either protons or deutrons. PET cyclotron can be self-shielded or unshielded and in the latter case, the cyclotron is installed inside a concrete vault room to prevent workers from receiving a dose from neutron (Mendez *et.al. 2005,* Mendez *et.al. 2004,* Hertel *et.al. 2004*).

Cyclone 18/9 is a type of cyclotron produced by *Ion Beam Applications* (IBA), Belgium which is a negative ion accelerator which can respectively accelerate hydrogen and deuterium ions to energy of 18 MeV and 9 MeV. Negative ions are extracted using a foil stripper so that electrons from negative hydrogen ions or electrons from accelerated negative deuterium ions will be released to form protons or deuterons which are then fired into the target material. The ion beam current that can be achieved on the Cyclone 18/9 foil stripper is 80 μA for protons and 35 μA for deuterons respectively (Kusuma *et.al. 2012, IBA. 2009,* Suryanto*. 2005*). The cyclotron owned by *Muchtar Ryadi Comprenensive Cancer Center Siloam* (MRCCC Siloam) is Cyclone 18/9, but when this research was carried out, the cyclotron has been upgraded into negative ion accelerator so that both of them generate 18 MeV protons to produce ^{18}F.

In a cyclotron, accelerated charge particles interaction with a target or materials of the component produces x-ray, gamma, and neutron (Vega-Carrillo, *et.al*. 2006). Exposure of neutron and gamma due to cyclotron operation will increase the risk of radiation hazard to patients and the public (Mukherjee, 2004). Gamma and neutron radiation must be controlled to protect workers as well as the public from emerging radiation hazards. Radiation measurement should be carried out as a part of radiation protection monitoring program.

Refering to the provision in the Chairman of BAPETEN (Indonesian Nuclear Regulatory Body) Regulation No. 4 of 2013 Article 25 point b concerning the obligation of the Permit Holder to carry out radiation monitoring, the level and type of radiation outside the cyclotron radiation shielding have to be measured and be known its dose because it is related to the safety to workers who are in the radiation field. There are two types of radiation produced in the operation of the cyclotron, namely gamma and neutron radiation

In this study, measurement of gamma dose, neutron dose and its spectrum were carried out on the outer walls of the 18 MeV cyclotron radiation shielding. The expected results in this study were the ambient dose of gamma and neutron and also the neutron spectrum on the outer wall of the cyclotron radiation shielding when it is operated. The measurements of gamma and neutron dose will be compared with the measurements in the 18 MeV Cyclone IBA cyclotron with 2 metres thick shielding in other places, namely in IPEN, Brazil (Silva, *et.al.* 2011) and AUBMC, Lebanon (Al-Kattar, *et. al.* 2015).

MATERIALS AND METHODS

Materials

Dosimeter OSL

Dosimeter OSL is a dosimeter that utilizing light to stimulate dose information stored in dosimeter material. OSL dosimeter which is made from Al_2O_3:C is able to detect photon and beta. To detect neutrons, Al_2O_3:C is coated with $^6Li_2CO_3$. Compared with TL dosimeter, these dosimeter is more sensitive to low dose, simpler, high precision and accuracy, fast reading process, dose information stored in dosimeters can be read again, and dose can be accumulated against previous dose (Musa *et.al. 2017,* Mckeever *et.al. 2003*). An environmental OSL dosimeter produced by Landauer has 4 elements Al_2O_3:C as shown in **FIGURE 1**.

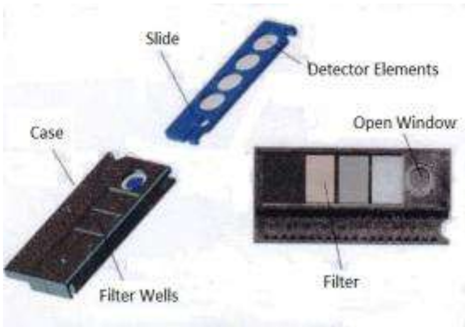


FIGURE 1. InLight environmental dosimeter

Dose information is able to be read by a *microStar reader* that is designed to be operated manually. *MicroStar reader* can be used to read dose in the laboratory and in the field.

Bonner sphere spectrometer

Bonner sphere spectrometer (BSS) is a neutron detector mounted in the center of a neutron moderating polyethylene sphere of various diameters. From the measurements, information can be derived to the spectrum of the neutron field where measurements were carried out (Tursinah *et.al. 2017,* Rodrigues *et.al. 2014*). A Bonner sphere spectrometer (BSS) is used in radiation protection measurement because of its wide energy range (thermal to tens MeV) and its easy operation (Tursinah *et.al. 2017,* Ogata *et.al. 2011*). In this study, BSS was used with a ^{6}LiI(Eu) detector which had 7 balls of polyethylene with diameters of 0, 2", 3", 5", 8", 10" and 12" respectively. The Bonner sphere spectrometer is shown in **FIGURE 2**.

FIGURE 2. Bonner sphere spectrometer at Neutron Laboratory of National Nuclear Energy Agency (NNEA)

UMG 3.3 Program

The UMG 3.3 program is an unfolding program issued by the German Physikalisch Technische Bundesanstalt (PTB) in 2004 (Reginato. 2004). With UMG program 3.3, BSS count rate values without moderator, moderated by 2", 3", 5", 8", 10 "and 12" can be converted to spectrum and the dose rate is calculated. The dose and spectrum of neutron is obtained from the BSS measurements which were unfolded using the Unfolding Maxed and Gravel (UMG) program. To run the UMG program, several inputs are required which are BSS count rates, BSS matrix responses, and reference spectrums. Matrix responses for BSS with LiI(Eu) detector have been calculated using the MCNPX program by Rasito *et al* (Tursinah *et al*. 2017)

Methods

Gamma Measurement

Four environmental OSL dosimeters were prepared to measures gamma doses. Three dosimeters are used to measure ambient dose of gamma, while one dosimeter is the control. All dosimeters were annealed and their reading was confirmed to be zero. All dosimeters were wrapped in aluminum foil to prevent Al_2O_3:C element from being exposed to light. Each dosimeter is labeled and placed in a plastic pocket and mounted on three points of the outer wall of 18 MeV IBA Cyclotron shielding, namely the bunker entry wall, the power supply wall, and the general store wall. The first two points are those which are close to the target chamber. The measurement points are shown in **FIGURE 3**.

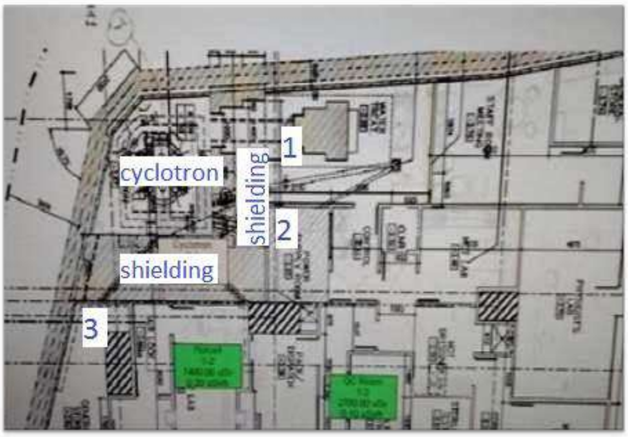


FIGURE 3. Points of gamma measurement are bunker entry wall (1), power supply wall (2), and general store wall (3).

After three months, dosimeters are taken and ambient dose is read using *microStar reader*.

Neutron Measurement

BSS and 6 balls of polyethylene diameter of 2", 3", 5", 8", 10" and 12" with their supporting accessories are prepared. BSS device and its balls are installed on the outer wall of bunker entry and general store. It was determined that neutron counts for each counting is twenty five counts to adjust the cyclotron operating time per cycle. Neutron measurement begin with polyethylene balls of 12", 10", 8 ", 5", 3", 2" and 0 (bare).

The counting value and its time are used as input to calculate ambient dose rate and to determine neutron spectrum using UMG 3.3 program.

RESULTS AND DISCUSSION

Gamma Ambient Dose

Gamma ambient dose on the outer wall of the cyclotron shielding is shown in **TABLE 1**.

Gamma ambient dose which was detected on the general store wall is predicted to originate from ^{18}F exposure when ^{18}F was transferred from the cyclotron target chamber to the FDG Hot Lab. When ^{18}F is transferred from the target chamber to the FDG Hot Lab, it was passed in the path outside of the shielding which is not as thick as the shielding near the general store wall. Although only in the order of second, high gamma radiation of ^{18}F is captured by an OSL dosimeter mounted on a general store measurement point. Comparing the measured ambient dose on the outer wall of the general store, the bunker entry and the power supplyl, it can be suggested that the ambient dose of gamma exposure due to the operation of the 18 MeV cyclotron was not detected. Concrete shielding wall is able to withstand gamma radiation emitted by cyclotron.

TABLE 1. Gamma ambient dose on the 18 MeV cyclotron shielding at MRCCC Siloam

No.	**Dose H * (10) (µSv/3 months)**	**Location**
1	ND	Bunker Entry Wall
2	ND	Power Supply Room Wall
3	367 ± 10.6 %	General Store Room Wall

Note: ND = not detected

With a thickness of 220 cm concrete shielding construction, gamma radiation produced can be absorbed by the shielding. Previously *Silva et.al.* (2011) measured gamma ambient dose rate at similar location of the Cyclone-18 MeV shielding wall using Geiger Muller Automes 6150 AD5 in 2010 and 2011. In this measurement, it was obtained the average gamma dose rate of (0.72 ± 0.24) □Sv/h and (0.63 ± 0.16) □Sv/h at cave access door. *Silva et al.* detect gamma radiation at the similar point to the point measurement at Cyclotron 18 MeV (Cyclone 18), MRCCC. Different measuring device gives different dose response where the Geiger Muller Automes 6150 provides a greater dose response. The transfer path of ^{18}F from target chamber to FDG Hot Lab might not be protected with adequate shielding and not far from the meaurement point. It can contribute to the dose. Meanwhile, the design of the ^{18}F transfer line on the cyclotron 18 MeV at the MRCCC is far from the area of the bunker entry wall and the power suppy room wall.

Neutron Ambient Dose Rate

Calculation result using UMG 3.3 program against neutron measurement data can be shown in **TABLE 2**.

TABLE 2. Ambient dose rate of neutron on the 18 MeV cyclotron shielding, MRCCC Siloam

No.	**Dose H *(10) of Neutron (□Sv/h)**			**Location**
	Fast	**Epithermal**	**Thermal**	
1	0.475± 12%	0.002± 23%	0.009 ±12%	Bunker Entry Wall
2	0.331± 12%	0.002± 16%	0.013 ± 9%	General Store Room Wall

The table shows that the largest ambient dose rate comes from fast neutron as of (0.475 ± 12%) □Sv/h and (0.331 ± 12%) □Sv/h on the outside wall of the bunker entry and the general store respectively. Meanwhile, ambient dose rate of thermal and epithermal neutron is relatively small. The ambient dose rate indicate that there are neutron leaks to outside of the concrete shielding although the value is very small. *Silva et.al.* measured neutron to outside of the Cylone18 MeV wall using Ludlum Model 15 at the similar points. In 2010, it was obtained neutron average dose rate of (3.6 ± 1.2) µSv/h on the cave access door. Meanwhile, in 2011, the ambient dose rate was (2.6 ± 1.3) µSv/h same point measurement. Measurement by *Silva et.al.* shows a greater total ambient dose rate compared to measurements in the similar measurement point on the outer walls of the 18 MeV cyclotron shielding by using a bonner sphere spectrometer. It shows that a different measuring device responds to different ambient dose rate as stated by *Kashougi et al.* (2015). However, neutron measurements using a bonner sphere spectrometer have the advantage of being able to measure fast neutron, eipithermal neutron, and thermal neutron, and also their energy spectrum.

Gamma and neutron dose rate is affected by energy and beam current of the cyclotron particles, material and thickness of the shielding, and also type of the target. In safety perspective, protons with energy of 18 MeV that produce neutron dose rate of 0.475 µSv/h on the outer wall of the concrete shielding with more than 2 meters are acceptable. When compared with the simulation and measurement in other place, the result is not much different.

Total neutron and gamma dose rate at the Cyclone 18 MeV with beam current of 100 µA with 2 meters of concrete shielding was 0.47 µSv/h (Al Kattar, *et.al.* 2015).

Neutron spectrum calculated using UMG 3.3 program can be seen in **FIGURE 4** and **FIGURE 5**.

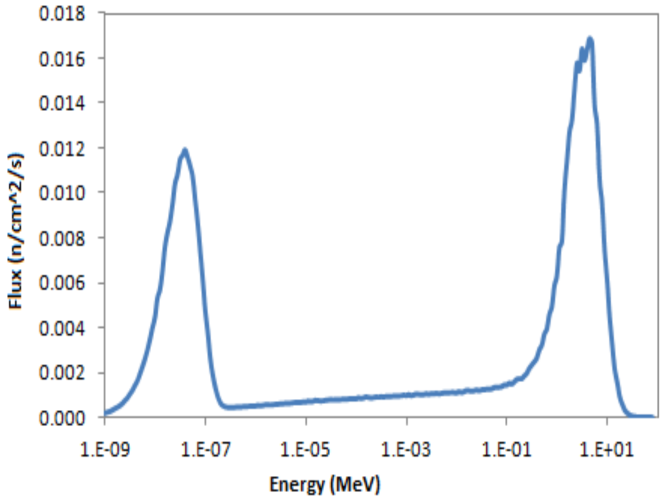


FIGURE 4. Neutron spectrum on the bunker entry wall of 18 MeV cyclotron shielding MRCCC Siloam

Referring to the neutron classification based on its energy, the dominant neutron flux is energy > $1x10^{-2}$ MeV (fast neutrons) and it was followed by < $5x10^{-7}$ MeV (thermal neutron).

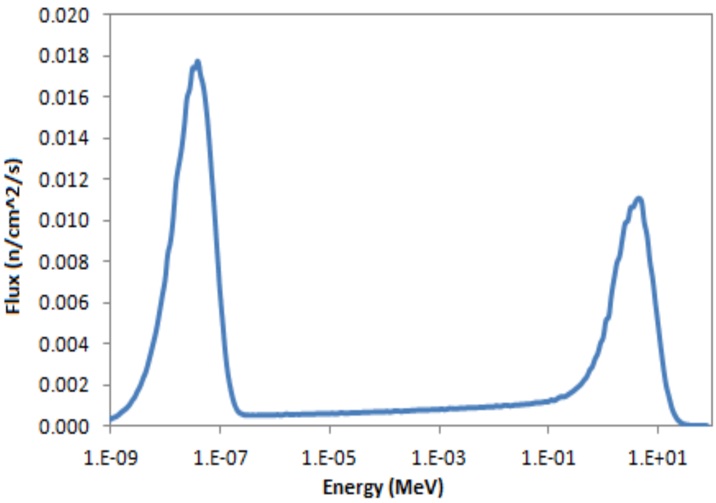


FIGURE 5. Neutron spectrum on the general store wall of 18 MeV cyclotron shielding MRCCC Siloam

CONCLUSION

Measurement of gamma ambient dose on the outside of the 18 MeV IBA cyclotron shielding wall at MRCCC Siloam provides an illustration that gamma radiation originating from cyclotron operation can be absorbed by 220 cm thick concrete shielding. The ambient dose of gamma which was detected on the outer wall of the general store is thought to originate from ^{18}F exposure when ^{18}F transferred from the cyclotron target chamber to the FDG Hot Lab. Measurement using other type of gamma radiation measuring device provides a different response. The type of gamma measuring device and the ^{18}F transfer line to the FDG Hot Lab that is not adequately shielded can cause a greater ambient dose response.

Neutron ambient dose rate which was detected on the outer shielding wall of the bunker entry and the general store indicates presence of neutron leak which is dominated by fast neutron. Meanwhile, ambient dose rate of thermal and epithermal neutron is relatively very small. Calculation using UMG 3.3 program generates a neutron spectrum and total average ambient dose rate < 0.5 Sv/h. A different neutron measuring device responds to different ambient dose rate, however the bonner sphere spectrometer has the advantage of being able to measure fast neutron, eipithermal neutron, thermal neutron and their energy spectrum.

ACKNOWLEDGMENT

The authors would like to thank to the Center for Technology of Radiation Safety and Metrology – NNEA for funding support of this research and Mr. Mohammad Faruq, Head of the Cyclotron Facility, MRCCC Siloam who provided the opportunity to conduct the research activities.

REFERENCES

[1] Anonim. (2013). *Proteksi dan Keselamatan Radiasi dalam Pemanfaatan Tenaga Nuklir.* Peraturan Kepala BAPETEN N. 4 Tahun 2013.

[2] Hertel N.E., Shannon M.P., Wang Z.I., Valenzano M.P., Mengesha W., and Crowe R.J. (2004). *Neutron Measurement in the Vicinity of a Shelf-shielded PET Cyclotron.* Radiation Protection Dosimetry 108 : 255-261.

[3] Kashougi M.A., Noor J.A.E., and Bunawas. (2015). *Tanggapan Surveimeter Neutron Terhadap Spektrum Campuran Energy Neutron (Response of Neutron Surveimeters Againts The Spectrum Energy Mixture Neutron).* Prosiding Seminar Nasional Keselamatan, Kesehatan, Lingkungan dan Pengembangan Teknologi Nuklir I; 40-46

[4] Kusuma A., Tuloh R.A., dan Suryanto H.(2012). *Pengoperasian Cyclone 18/9 untuk Produksi Radionuklida ^{18}F dalam Penyiapan FDG di Rumah Sakit MRCCC Jakarta*. Prosiding Pertemuan dan Presentasi Ilmiah Teknologi Akselerator dan Aplikasinya Vol 14: 202-212.

[5] Landauer Inc. (2009). *Dosimeter Designation: InLight Environmental Dosimeter.* Retrieved Juni 10, 2019 from https://www.nagase-landauer.co.jp/english/inlight/pdf/Dosimeters/ einvironmentaldosimeters.pdf

[6] Landauer Inc. (2009). *Landauer,s Albedo Neutron Detector.* Retrieved Juni 10,2019 from https:// www.nagase-landauer.co.jp/ english/inlight/ pdf/Dosimeters/neutrondosime-ters.pdf

[7] McKeever S.W.S. and Moscovitch M. (2003). *On the Advantages and Disadvantages of Optically Stimulated Luminescence Dosimetry and Thermoluminescence Dosimetry.* Radiation Protection Dosimetry 104 (3): 263-270.

[8] Mendez R., Inigues M.P, Marti-Climent J.R., Penuelas I., Vega-Carrillo H.R., and Barquero R. (2005). *Study of the Neutron Field in the Vicinity of an Unshielded PET Cyclotron.* Physics Medicine Biology 50: 5141-5152.

[9] Mendez R., Marti-Climent J.R., Inigues M.P, Vega-Carrillo H.R., Barquero R. , and Penuelas I. (2004). *Study of the Neutron Field around a PET Cyclotron.* IRPA 11

[10] Mukherjee B. (2002). *Principle of Radiological Shielding of Medical Cyclotrons*, Proceedings of the Sixth Meeting of the Task Force on Shielding Aspects of Accelerators, Targets and Irradiation Facilities.

[11] Musa Y., Hashim S., Karim M.K.A., Bakar K.A., Ang W.C., and Salehhon N. (2017). *Response of Optically Stimulated Luminescence Dosimeters Subjected to X-rays in Diagnostic Energy Range*. Journal of Physics: Conference Series 851 012001.

[12] Ogata T., Kudo S., Watanabe Y., Muramatsu T., Yamamoto H., Iwai S, Tagaki S., Harano H., Matsumoto T., and Nishiyama J. (2011). *The Calibration of Bonner Sphere Spectrometer.* Radiation Protection Dosimetry Vol. 146 No.1-3:107-110.

[13] Ortiz-Rodriguez J.M., Reyes Alfaro A., Reyes Haro A., Cervantes Viramontes J.M., and Vega-Carrillo H.R. (2014). *A Neutron Spectrum Unfolding Computer Code Based on Artificial Neural Network.* Radiation Physics Chemistry Vol. 95: 428-431.

[14] Reginatto M., UMG package, version 3.3 - release date: March 1, 2004.

[15] Silva P.P.N., Fernandes I.M., da Silva A.J., Rodrigues D.L., and Romero Filho C.R. (2011). *Neutron and Gamma Radiation Levels Analysis for 18 MeV Cyclotron Operation at IPEN-CNEN-SP.* Proceeding INAC 2011.

[16] Suryanto H. (2005). *Teori Operasi Siklotron.* Manual Coaching Pusdiklat-BATAN.

[17] Tursinah R., Bunawas., *and* Kim J. (2017). *Neutron Response Function of a Bonner Sphere Spectrometer with ^{6}LiI(Eu) Detector.* Ganendra Journal of Nuclear Science and Technology Vol. 20 No. 2: 65-72.

[18] Vega-Carrillo H.R., Mendez R., Iniguez M.P., Climent J.M., and Penuelas I. (2006). *Neutron Field Inside a PET Cyclotron Vault Room*. Primer Congreso Americano del IRPA 2006 /First American IRPA Congress 2006 XXIV Reunión Anual de la SMSR y XVII Congreso Anual de la SNM/XXIV SMSR Annual Meeting XVII Annual SNM Congress Acapulco México, del 4 al 8 de Septiembre 2006.

[19] Al Kattar Z., El Balaa H., Nassredine M., and Haydar M. (2015). *Radiation Safety Issues Relevant To Radioisotope Production Medical Cyclotron*. International Conference on Advances in Biomedical Engineering (ICABME).

Study Review of Nuclear Medicine Facility Safety Procedure During COVID-19 Pandemic

Anita Nur Mayani[1, a)]

[1]D3 Radiology Course Program, STIKES Guna Bangsa Yogyakarta, Sleman, Yogyakarta 55283, Indonesia

[a] anita.nur81@ui.ac.id

Abstract. Nuclear medicine facility provides therapeutic and diagnostic (theranostic) services. In nuclear medicine services, patients have been scheduled for treatment. During the COVID-19 pandemic, the patient does not allow to visit the hospital except in urgent condition by following COVID-19 protocols. Since it would be high risk for patient with non-communicable disease which suffering by nuclear medicine patient. Nuclear medicine facility procedures in Indonesia following hospital COVID-19 protocol. For this reason, the author did a study review about nuclear medicine facility safety during COVID-19 pandemic. The literature review study was conducted referring to the WHO international guidelines and specific nuclear medicine IAEA. To map out the nuclear medicine facility procedure during COVID-19, the author compiling nuclear medicine guidelines and COVID-19 safety guidelines. The procedures consisted of the safety procedures of healthcare workers, patients, and disinfecting areas of nuclear medicine facilities. This paper is expected to be a safety reference in helping policymakers Indonesia for nuclear medicine practitioners. In addition to creating a safe work environment habit in COVID-19 pandemic by minimizing the risk of virus spreading.

Keywords: Nuclear Medicine Facility, Safety procedure, COVID-19.

INTRODUCTION

Since March 2020 WHO announced that Corona Virus Disease (COVID-19) has become a global pandemic. People fight against COVID-19 to minimize the spreading of the virus. Researcher conducted to help deal with pandemic until the discovery of COVID-19 vaccine. People survive by adapting the new normal that is applied in life during pandemic as follows physical distancing, wearing the mask, and sanitizing. World Health Organization has developed rules to be applied during the pandemic in public space especially hospitals [1]. One of the main health services in hospitals is nuclear medicine facility. In the field of nuclear medicine, the specific procedures applied during a pandemic have been published by the IAEA in July 2020. [2]

Nuclear medicine facilities provide diagnostic services in vivo, in vitro, and therapy services. Diagnostic in vivo in nuclear medicine provides bone uptake, thyroid uptake, Sestambi Stress Test Myocardial Perfusion Image (SST MIBI), Glomerular Filtration Rate (GFR), Renography, liver uptake, MUGA, and lung perfusion [3]. Diagnostic in vitro in nuclear medicine provides diagnostic examination by taking samples of patients to be tested in the nuclear medicine laboratory. Nuclear medicine therapy is conducted for cancer patients [4]. The disease that is performed in nuclear medicine facility is mostly non-communicable disease with a high risk of exposure by COVID-19 [5]. This is caused by the condition of the body that is not good and affects the body's immune system [6]. COVID-19 can negatively impact Non-Communicable Disease (NCD) outcomes for adults and children through several pathways including the higher susceptibility to COVID-19 infection and higher case fatality rates among people with NCDs; For example, COVID-19 has been associated with cardiovascular complications that can make the accurate diagnosis of myocardial infarction more difficult. In addition, patients with chronic respiratory diseases face particular challenges in making choices about when to seek care, since their baseline disease may cause signs and symptoms similar to those of COVID-19. Cancer treatment plans should consider the increased morbidity and mortality caused by COVID-19 in cancer patients, and multidisciplinary teams can support the definition of priority interventions [7].

Therefore nuclear medicine facilities need to apply COVID-19 procedure specifically to minimize COVID-19 spreading. Besides the safety of health workers and patient are the most important thing which has to be considered [4]. During COVID-19 pandemic, safety in nuclear medicine not only considers the radiation but also in infectious COVID-19 spreading. Therefore it is important to adjust procedures related to these aspects. Patient workflow is an essential thing to be considered. Starting from patient triage, treatment patient until the patient finished the treatment. The nuclear medicine facility is divided into three main areas including radioisotope room, patient areas, and administrative areas of officers [4]. Then safety aspects of nuclear medicine facilities are building construction and medical devices. Nuclear medicine facilities should apply administrative control and environmental and engineering controls to adopted the COVID-19 safety procedure [8]. It is can be done by prioritizing worker safety and support, patient service delivery, data streams for situational awareness, facility practice, and communications [9].

Nuclear medicine in Indonesia using COVID-19 protocols in general. Whereas nuclear medicine patients have been scheduled for treatment based on previous treatment in some cases. It is quietly different from general practitioner control. Therefore the author is interested to do a study review to map out the nuclear medicine facility procedure during COVID-19 by compiling nuclear medicine guidelines and COVID-19 safety guidelines. The results of this study are expected as a reference for policymakers also healthcare workers in a nuclear medicine facility to provide a safety environment during pandemic COVID-19.

METHOD

This paper was prepared using a literature review by conducting conceptual studies relating to the safety of nuclear medicine facility during pandemic COVID-19. From the conceptual result of the studies by document review and then the descriptive analysis is done by collaborating radiation safety procedures and COVID-19 safety procedures. Afterward, the result procedure of nuclear medicine facility during pandemic COVID-19 is compiled from documents guideline, papers, and research references. The literature review study was conducted referring to the WHO international guidelines and specific nuclear medicine IAEA.

RESULT AND DISCUSSION

Patient Service Delivery Safety Procedure

Nuclear medicine is a medical science that utilizes radioisotope to carry out diagnostic and therapy (theranostic). Nuclear medicine can see the physiology of the human body by injecting radiopharmaceutical into the vein. Nuclear medicine treatment is different from other medical treatment as it uses radioisotopes that emit radiation. This could be dangerous for patients and the environment if not according to safety procedures. Therefore safety the most important in nuclear medicine facility. Nuclear medicine facilities in general consist of administration room, waiting room, examination room, laboratory, scanning room, waiting dose room, and consultation room. The workflow of nuclear medicine patient starts from administration room – consultation room – examination room – scanner room – waiting dose room.

The general principles that must be applied during the COVID-19 pandemic in the nuclear medicine department including: (1) Distancing in nuclear medicine department at least 1 meter; (2) Hand hygiene with water and soap or if not available can use hand sanitizer that contains 60% alcohol or more frequently; (3) Rescheduling non-urgent procedures, (4) Ensuring supplies PPE are available for staffs; and also (5) Promoting using telehealth. [2] In general the procedures that patient has to be conducted including: Maintain physical distance of at least 1 meter, provide medical mask if tolerated by patient, perform hand hygiene and have the patient perform hand hygiene. For patients who have COVID-19 symptoms, staff immediately move the patient to an isolation room or separate area away from others. **TABLE 1** explains the flow of patient's radiation safety that be compiled with COVID-19 safety procedures.

TABLE 1. Patient procedure in nuclear medicine facility during pandemic COVID-19 [10]

Area	Procedure
1. Patient arrival – waiting room	● Thermal screening. ● Access to handwashing facilities and tissue boxes and masks are within easy reach. ● Enough space so that waiting patients may sit at enough distance, as the risk of transmission increases within three feet. ● When such patients are identified they should be placed in a separate waiting area.
2. During uptake phase – patient waiting room for PET	● Maintain physical distance of at least 1 meter ● Use mask and eye protection ● Stay hand hygiene
3. While the patient scanned and goes home	● Stay hand hygiene

Worker Safety Procedure

Healthcare workers in the nuclear medicine facility consist of medical specialists, radio pharmacist, radiographer, and nurse. Nuclear medicine frontline staff such as radiographers or nurses will have the most potential close contact with infected patients in pandemic COVID-19 [9]. So the healthcare workers need to be more careful about the safety aspect of radiation and virus exposure. Based on current evidence, the COVID-19 virus is transmitted between people through close contact and droplets. Airborne transmission occurs during aerosol-generating procedures and support treatments. Infection prevention and control strategies in health care to prevent or limit COVID-19 transmission including ensuring triage, applying standard precaution of diligent hand hygiene, implementing empiric additional precautions, administrative control, and using environmental and engineering controls. **TABLE 2** describes the procedures for workers for safety in nuclear medicine facility areas.

TABLE 2. Healthcare workers safety procedure in nuclear medicine facility [2], [10]

Area	Workers
Screening/triage Administrative area	• Maintain physical distance of at least 1 meter. • Ideally, build glass/plastic screens to create a barrier between health care workers and patients. • No PPE is required. • When the physical distance is not feasible and yet no patient contact, use mask and eye protection.
Consultation room	• Medical mask • Gown • Gloves • Eye protection • Perform hand hygiene
Injection room	• Use all aseptic and antiseptic techniques. • Apply all standard radiation protection and optimization principles. • Use the appropriate PPE. • Place special attention when removing the gloves and other protective elements. • Disinfect the devices used during patient preparation and injection. • Thoroughly sanitize hands after each procedure. • Dispose of the used protective elements in a container for biosafety waste
Scanning room	• Apply all standard radiation protection and optimization Principles. • Use the appropriate PPE. • Use disposable protective elements for the scanners.
Laboratory	• Maintain physical distance of at least 1 meter • Medical mask • Eye protection • Gown • Gloves • Perform hand hygiene
Patient room	• Respirator N95 or FFP2 or FFP3 standard, or equivalent. • Gown • Gloves • Eye protection • Apron • Perform hand hygiene

The same precautions and screening that apply to the patients on arrival should in theory apply to nuclear medicine staff (e.g., technologist, nurses, nuclear medicine physicians, and radiologist). Simple measures such as staying home if unwell and particularly if having traveled to known COVID-19 affected countries would do much to reduce the risk of virus transmission. We would suggest that senior clinicians and/or management take a more proactive stance to advise staff that they should not come to work if they are not well.

Nuclear Medicine Decontamination Procedure

Nuclear medicine facility building has been designed to be easily disinfected from the radioisotope. This makes it easy to disinfect the surface that has been touched frequently. In some rooms such as dose waiting rooms that at risk of contamination are designed with negative pressure. Unfortunately, not all facilities apply it correctly. The floor and ceiling walls with solid structures are required and easily cleaned from contamination [3]. Optimized techniques may minimize, but not eliminate, a small degree of airborne contamination. Besides, patients frequently cough following inhalation of the radiopharmaceutical, which may also expose nuclear medicine workers to aerosolized secretions [11]. WHO classified decontamination in medical devices by risk category (Spaulding classification) including High (critical) that need decontamination in level sterilization, intermediate (semi-critical) – disinfection (high level), and low (non-critical) – cleaning (visibly clean). However, this classification adjusted to the maintenance instructions from the manufacturer [12].

TABLE 3. Decontamination procedures in nuclear medicine facility during COVID-19 pandemic [2], [10], [13], [14]

Room area	Frequency	Additional guidance
Screening/triage, Waiting room, Consultation room, Administrative area.	At least twice daily	• Focus on high-touch surfaces, then floors (last)

Scanner room	After each patient visit (in particular for high-touch surfaces) and at least once daily terminal clean	• High-touch surfaces to be disinfected after each patient visit • Once-daily low-touch surfaces, high-touch surfaces, floors (in that order); waste and linens removed, examination bed thoroughly cleaned and disinfected
Patient room	At least twice daily	• High-touch surfaces to be disinfected after each patient visit • Once-daily low-touch surfaces, high-touch surfaces, floors (in that order); waste and linens removed, examination bed thoroughly cleaned and disinfected
Shared toilet	At least three times daily	• High-touch surfaces, including door handles, light switches, counters, faucets, then sink bowls, then toilets, and finally floor (in that order) • Avoid sharing toilets between staff and patients

CONCLUSION

Resume nuclear medicine procedure in COVID-19 pandemic in **FIGURE 1** by applied general principles recommendation from IAEA during the COVID-19 pandemic in the nuclear medicine department. Safety in nuclear medicine facility is very important especially for patient which most of with high-risk COVID-19 comorbidities. Therefore nuclear medicine facility procedures must be concerned with virus contamination while prioritizing radiation safety for patient and health workers. The structure of nuclear medicine building has been designed easy to decontaminate. However diligent in hygiene life behavior is important as COVID-19 is a new virus that continues to be researched.

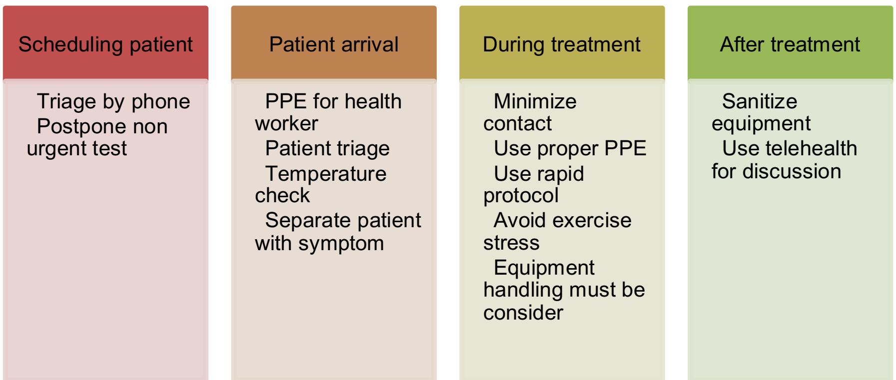


FIGURE 1. Nuclear medicine procedure resume in COVID-19 pandemic (author)

REFERENCES

[1] World Health Organization, “Strategic Preparedness and response plan” (World Health Organization, Geneva, 2020).

[2] International Atomic Energy Agency, “COVID-19 Pandemic: Technical Guidance For Nuclear Medicine Departments IAEA” (IAEA, Austria, 2020).

[3] Rumah Sakit Hasan Sadikin, “Kedokteran Nuklir”. [Online] Available: http://web.rshs.or.id/. [Accessed: 5-July-2020].

[4] Mayani, AN and Basari, "Evaluating on requirements of nuclear medicine facility building" (AIP Conference Proceedings 2180, 020049 (2019): https://doi.org/10.1063/1.5135558 Published Online: 10 December 2019)

[5] Centers for Disease Control and Prevention, “Coronavirus Disease 2019 People with Certain Medical Conditions” [Online] Available: https://www.cdc.gov/coronavirus/2019-ncov/need-extra-precautions/people-with-medical-conditions.html. [Accessed: 5-July-2020].

[6] Kementerian Kesehatan Republik Indonesia, “Orang dengan Penyakit Tidak Menular Beresiko Tinggi Terinfeksi COVID-10” [Online] Available: https://www.kemkes.go.id/article/view/20070500003/orang-dengan-penyakit-tidak-menular-berisiko-tinggi-terinfeksi-covid-19.html. [Accessed: 5-July-2020].

[7] World Health Organization, “Maintaining essential health services: new operational guidance for the COVID-19 context” [Online] Available: https://www.who.int/news-room/detail/01-06-2020-maintaining-essential-health-services-new-operational-guidance-for-the-covid-19-context. [Accessed: 30-June-2020].

[8] World Health Organization, “Rational use of personal protective equipment for coronavirus disease (COVID-19) and considerations during severe shortages” (World Health Organization, Geneva, 2020).

[9] Centers for Disease Control and Prevention, “Coronavirus Disease 2019 Operations for Healthcare System” [Online] Available: Centers for Disease Control and Prevention, “Coronavirus Disease 2019 People with Certain Medical Conditions” [Online] Available: https://www.cdc.gov/coronavirus/2019-ncov/need-extra-precautions/people-with-medical-conditions.html. [Accessed: 5-July-2020].

[10] Skali, H. et al. “Guidance and Best Practices for Nuclear Cardiology Laboratories during the Coronavirus Disease 2019 (COVID-19) Pandemic: An Information Statement from ASNC and SNMMI” [online] Available: https://www.sfmn.org/drive/CNP/CODIV-19/COVID-19%20ASNC%20SNMMI%20Preprint.pdf. [Accessed: 5-September-2020].

[11] Huang, Hian Liang et al. “COVID19 -Nuclear Medicine Departments, be prepared!.” Nuclear medicine communications vol. 41,4 (2020): 297-299. doi:10.1097/MNM.0000000000001183

[12] Zuckier, Lionel S.; Gordon, Sharon R. “COVID-19 in the Nuclear Medicine Department, be prepared for ventilation scans as well!.” Nuclear Medicine Communications: May 2020 - Volume 41 - Issue 5 - p 494-495 doi: 10.1097/MNM.0000000000001196

[13] World Health Organization, “Decontamination and Reprocessing of Medical Devices for Healthcare Facilities” (World Health Organization, Geneva, 2016).

[14] World Health Organization, “Cleaning and disinfection of environmental surfaces in the context of COVID-19” (World Health Organization, Geneva, 2020).

Performance Assessment of Health Facility Based On Implementation Of Radiation Protection Optimization On Occupational Exposure: A Review

Endang Kunarsih [1, a)]

[1] *Badan Pengawas Tenaga Nuklir, Jl. Gajah Mada No. 8 Jakarta Pusat, Indonesia*

[a)] e.kunarsih@bapeten.go.id

Abstract. Radiation protection optimization for occupational exposure is a radiation protection and safety process carried out as an effort to control that the radiation dose received by radiation workers during their duties can be as low as possible which can be achieved by considering economic and social factors. The implementation of the optimization principle is unique for each health facility, and impact the performance of the facility in terms of radiation protection and safety. Currently, there is no comprehensive formula to assess the performance of health facilities in implementing the optimization of radiation protection in their activities. A quantitative assessment will be proposed in this paper through a descriptive review based on some literature from the IAEA and regulations. The proposed method uses a rating scale for the assessment parameters based on a management perspective and a technical perspective with the score acceptance criteria adopting the fulfillment of the PDCA principles. The quantification of the assessment model is intended to make it easier to monitor the level of achievement of radiation protection optimization performance on occupational exposure and determine the appropriate improvement strategy, both for regulators and the licensee.

Keywords: performance assessment, radiation protection optimization, occupational exposure

INTRODUCTION

The implementation of radiation protection and safety optimization principle is unique for each health facility. It can be the strength and weakness that affect a facility's performance in terms of radiation protection and safety. Safety culture will be reflected in every action in various aspects of implementing optimization. Resources will be taken into account in taking action for optimization efforts. The implementation of optimization at the operational stage in health facilities in addition to fulfilling the mandate of regulations also shows the ability of health facilities in conducting good practices to seek optimal radiation protection for workers, patients, and the public.

In terms of facility performance evaluation, the parameters currently used by BAPETEN are still in the form of compliance with licensing requirements and operational requirements following regulations. These parameters are indicators of safety and/or security of facilities and are listed in the BAPETEN Chairman Regulation (BCR) Number 1 Year 2017 Article 55, which consists of licensing conditions, availability of human resources, monitoring of radiation doses, health assessment for radiation workers, availability of safety and security equipment, monitoring of occupational radiation exposure, and availability or suitability of documents and records [1].

It appears that there is no comprehensive formula to assess the performance of health facilities in implementing the principle of radiation protection optimization in their activities. An assessment of the quality of the implementation process has been published in Kunarsih, E (2019) with the title 'Strengthening the Protection of Radiation Workers in Health Facilities through Self-Assessment of the Effectiveness of Optimization of Radiation Protection on Occupational Exposure: A Review' [2], while quantitative assessment will be proposed in this paper through a descriptive review. The quantification of the assessment model is intended for facilities to easily monitor the level of achievement of radiation protection optimization performance on occupational exposure and determine the appropriate improvement strategy. The resulting value also can be used as supporting data for government regulators, both BAPETEN and the Ministry of Health (MOH) in the monitoring of health facilities performance.

THEORY

Occupational Exposure in A Health Facility

Occupational exposure is the exposure of workers incurred in the course of their work [3]–[5]. Radiation exposure received by workers can originate from activities as presented in **TABLE 1** [6], [7]:

TABLE 1. Activities that give rise to occupational exposure

Facility	Activities
Radiotherapy	- patient imaging - radiotherapy treatment - radiotherapy equipment and radioactive sources quality control - radioactive sources handling, storage, and replacement
Nuclear medicine	- patient imaging - receiving radioactive material from suppliers, - radioactive activity measurement, - radioactive substances storage - radionuclide elution - radionuclide labeling - radionuclides administration to patients - patients examination and treatment - radioactive substances internal transport - radioactive waste handling - radioactive contamination
Diagnostic and interventional radiology	- patient imaging - interventional radiological treatment - X-ray equipment quality control

Radiation protection for occupational exposure, in general, can be implemented by controlling external radiation exposure and internal radiation exposure. In GSR Part 3, it was stated that to control exposure in planned situations is to use the good design of facilities, adequate equipment, proper operating procedures, and conduct training for radiation workers. Dose management for radiation workers must be carried out and based on planned exposure situations [8].

Optimization of Radiation Protection and Safety on Occupational Exposure

Optimization of radiation protection in occupational exposure is a process of radiation protection and safety carried out to ensure that the radiation dose received by radiation workers during their duties can be as low as can be achieved by considering economic and social factors [3], [4], [8]. At the operational stage of activities in health facilities, there are practical tools that can be used as parameters in controlling the optimization of radiation protection against occupational exposure, namely dose constraint for radiation workers [9].

In implementing the principle of optimizing radiation protection in facilities, several aspects need to be considered. These aspects can be determined from the perspective of work management and specific technical operational, as summarized in **TABLE 2** [10].

TABLE 2. Aspects of implementing the principle of optimizing radiation protection

Managerial perspectives	Operational perspectives
● improving work plan design	● carry out quality control of equipment
● organizing training	● updating operational methods/procedures
● increasing competency of worker	● reduction of work time in the radiation area
● increasing awareness and involvement of workers and management	● limiting the number of workers in the radiation area
● implementing effective communication in the work environment	● reduction in the dose rate received by workers
	● thematic training

These elements can be parameters for analyzing the implementation of optimization of radiation protection and safety on occupational exposure as a follow-up to the results of the annual dose constraint study of workers in a facility.

Performance Assessment

Assessment is a process to give value [11]. Assessment can be defined as one or more processes to identify, collect, and prepare data used to evaluate the achievement of a particular process. Thus, performance assessment is as a process or activity carried out by individuals or groups in an organization to evaluate how the organization does its work by comparing the results of its work with a set of standards/criteria that have been made in a certain period that is used as a basis for assessment considerations.. As described in ISO 9004, top management should assess progress in achieving planned results against the mission, vision, policy strategies, and objectives, at all levels and in all relevant processes and functions in the organization. A measurement and analysis process should be used to monitor this progress, to gather and provide the information necessary for performance evaluations and effective decision making [12].

In terms of implementing the optimization of radiation protection, assessment involves elements of examination and judgment in the evaluation. Many factors influence the implementation process of radiation protection optimization, therefore this assessment cannot be completely separated from the involvement of intuition and consideration. The quantification of the assessment model through a mathematical formula will make it easier to monitor the level of achievement of radiation protection optimization performance on occupational exposure and determine the appropriate improvement strategy.

RESULTS AND DISCUSSION

Identification of Assessment Parameters and Acceptance Criteria

Based on the above theory, it is stated that to examine the implementation of optimization of radiation protection in a facility, the parameters identified from management perspectives and technical operational perspectives should be used. In this paper, the authors identify these parameters (presented in **FIGURE 1**) based on IAEA documents (SRS 21 and GSG 7) and their acceptance criteria based on regulation.

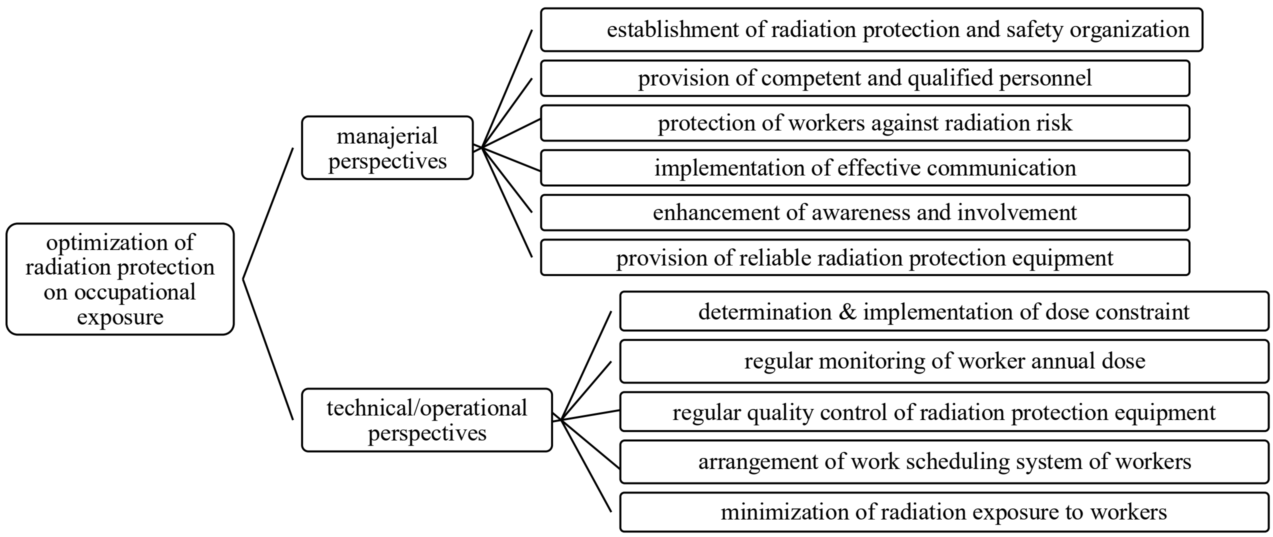


FIGURE 1. Implementation optimization parameters [10], [13].

In **FIGURE 1**, the managerial perspective consists of parameters that describe commitment, policy, and management's role in terms of fulfilling the principle of optimizing radiation protection for radiation workers.

a. Radiation Protection and Safety Organization (RPSO)

RPSO is important to be officially determined by the licensee, as a forum for organizing and controlling the radiation protection and safety system in a facility as mandated in Government Regulation (GR) Number 33 Year 2007 [4]. Therefore, the following parameters need to be confirmed:

- RPSO structure available, contained in official written documents and integrated into the overall organizational structure of the facility.
- Duties and responsibilities of personnel related to radiation protection and safety are clearly defined and detailed coordination paths are drawn in official written documents.

- Reviewing the RPSO structure is carried out periodically, taking into account changes in legislation and facility resources (personnel, budget, policies, etc.).
- A review of the Radiation Protection and Safety Program (RPSP) is conducted periodically, taking into account changes in legislation and facility resources (workers, budgets, policies, etc.), and changes in the condition of facilities (additional modalities and equipment).

b. Provision of competent and qualified radiation workers.

Personnel (radiation workers and other workers) who work in radiation facilities are one of the main keys to the proper implementation of radiation protection and safety. Therefore Licensee must determine the qualifications and competencies of its personnel and maintain and improve these competencies as mandated in Article 16 GR Number 33 Year 2007 [4]. Therefore, the following parameters need to be confirmed.

- Availability and documentation of management commitments related to the provision of competent and qualified workers including requirements for qualifications and competency of workers, analysis of training need to achieve and improve worker competence, and budget planning for worker training.
- Availability of routine training programs related to radiation protection and safety and special technical training following their fields of work, both internally and externally, both formally and in the form of routine coaching or briefings aimed at fostering workers' concern for protection and safety radiation.
- Reviews on the performance of workers and Radiation Protection Officer (RPO) are conducted periodically.

c. Protection of workers from the risk of radiation.

The licensee must monitor the radiation dose received by workers and monitor the health of workers as mandated by Article 25 GR Number 33 Year 2007 [4]. Therefore, the following parameters need to be confirmed.

- Availability and documentation of management commitments related to the protection of radiation workers through monitoring the dose and health of workers, including commitments to provide radiation protection equipment. This commitment can be implemented, among other things, through collaboration between Licensee and third parties related to the provider of worker dose evaluation services and worker health checks, and budget planning for the provision of individual dose monitoring equipment.
- Management's commitment to designing workloads and work scheduling systems in line with the principle of protecting workers from radiation risks.
- Management's commitment to immediately report and investigate if there is excessive radiation exposure to radiation workers (workers' doses exceed dose limit).

d. Implementation of effective communication.

Regular communication both formally and informally between all levels of management and workers is an important part of supporting the success of a system/process. The licensee needs to establish good communication networks at all levels of the organization, to produce an appropriate flow of information about radiation protection and safety [4]. Therefore, in this element it is necessary to confirm that communication patterns are available and implemented, both carried out vertically according to the bureaucratic hierarchy and horizontally between teams involved in radiological actions. Also, information systems or information support systems are adequately available for the effectiveness of the information dissemination process.

e. Enhanchment of awareness and involvement of radiation workers.

Direct involvement in the planning phase allows workers to apply experience and lessons that can be taken to develop the plans. So that potential risks can be identified and applied in developing plans for each job. Workers' involvement also in the process evaluation phase, post-work review, and feedback process will provide a lot of valuable information. Workers need to be convinced that their input is valued and can support the process of optimization [10]. Therefore, in this parameter, it is necessary to confirm that policies regarding the involvement of workers in planning, evaluating, and monitoring processes in facilities are available and documented. Also, the composition of the team and the role of each team member in each planning, evaluation, and monitoring activity at the facility is sufficient.

f. Provision of reliable radiation protection equipment.

Radiation protection equipment is important to supports the implementation of radiation protection, therefore Licensee must provide radiation protection equipment that functions properly according to the type of source and energy used as mandated by Article 31 GR Number 33 Year 2007 [4]. Therefore, in this parameter, it is necessary to confirm management's commitment about ensuring the quality and reliability of radiation protection equipment i.e radiation level monitoring equipment and/or work area contamination, individual dose monitoring equipment, apron, etc [5], and its supporting facilities such as software, hardware, and information technology. This commitment can be implemented in the form of cooperation between Licensee and third parties related to maintenance service providers, testing and/or calibration of radiation protection equipment, and budget planning for maintenance, testing, and/or calibration of radiation protection equipment.

In **FIGURE 1**, the technical perspective consists of parameters that describe the fulfillment of the principle of optimization of radiation protection in routine operational practices of the facility.

a. Determination and implementation of dose constraint for radiation workers at the operational stage.

In this parameter, it is necessary to confirm that the dose constraint for the radiation worker is established and reviewed and corrective action is taken for the condition of the annual dose of the worker which exceeds the dose constraint.

b. Periodic monitoring of workers' annual doses.

In this parameter, the following things need to be confirmed that:

- Compliance in the implementation of dose monitoring for radiation workers, keeping of radiation workers 'doses records (from passive dosimeters and active dosimeters), preparing dose cards for each radiation worker for a certain period, and keeping records of radiation workers' doses for certain activities.
- Awareness in the use and storage of individual dose monitoring equipment (TLD badge, TLD optic lens, TLD ring, and personal dosimeter) correctly.
- Compliance in the delivery of individual dose monitoring equipment to the dosimetry laboratory on time.
- Compliance in informing the results of monitoring the dose to each radiation worker.

c. Quality control of radiation protection equipment.

In this parameter, it is necessary to confirm that maintenance and/or quality control testing of radiation protection equipment (apron, shielding, etc.) and calibration of radiation level monitoring equipment (survey meter/dose rate measuring instrument, measuring instrument for surface contamination, and/or air contamination) and active personal dosimeter equipment is carried out routinely.

d. Arrangement of assignment/scheduling system for workers.

The number of workers involved in a task/job can be optimized as needed to complete the task without reducing the quality of the performance. Therefore this parameter needs to be confirmed that

- Assignment of workers in the area of radiation is sought at a minimum (in terms of number of people and working time) as needed for each type of work
- Workers whose types of tasks do not have a direct interaction to the source of radiation are not allowed to be in the radiation area.

e. Minimization of exposure to workers.

This parameter needs to be confirmed

- Design of radiological treatment rooms (control areas), waiting / isolation rooms of nuclear medicine patients, storage rooms for radioactive substances, waste storage rooms do not allow leakage.
- Infrastructure facilities for handling and storing radioactive sources and radioactive substances are fully available.
- The worker's position (orientation) towards the radiation source is appropriate in the context of reducing the rate of radiation exposure.
- The process of handling and storing radioactive sources and radioactive substances is carried out correctly.
- The minimization of contamination of workspaces, equipment, and workers is carried out correctly.
- Radiation exposure measurements in the work area and surrounding rooms are carried out routinely.
- Radiological reviews are carried out routinely to identify the need for changes to the boundaries of the work area and radiation protection and safety measures.

The Proposed Performance Assessment Model

The method proposed in this performance assessment is the rating scale, which is the use of a tiered numerical scale to measure the level of fulfillment of performance parameters [14]. The proposed score is from 1 to 5 with the scoring criteria adopted from the fulfillment of PDCA principles (Plan, Do, Check, Act) for a system/organization. The scoring is presented in **TABLE 3** as follows:

TABLE 3. Scoring criteria

Scoring	Criteria	
5	Plan	Policies, planning, program activities are established and documented.
	Do	Activities are carried out according to policies/plans/programs that have been set and records of the results of activities are controlled.
	Check	Implementation of activities are monitored / evaluated / reviewed
	Action	Corrective action is taken if unconformity is found during monitoring/evaluation/review

4	Plan	Policies, planning, program activities are established and documented.
	Do	Activities are carried out according to policies/plans/programs that have been set and records of the results of activities are controlled.
	Check	Implementation of activities are not monitored/evaluated/reviewed
	Action	Corrective action is only done partially because of findings from external parties (for example BAPETEN, KEMENKES, KARS, and others)
3	Plan	Policies, planning, program activities are established but not documented.
	Do	Activities are carried out as needed and records of the results of activities are controlled.
	Check	Implementation of activities are not monitored/evaluated/reviewed
	Action	Corrective action is only done partially because of findings from external parties.
2	Plan	Policy, planning, program activities are not established.
	Do	Activities are carried out as needed but the results of the recording are not controlled.
	Check	The implementation of activities is not monitored/evaluated/reviewed.
	Action	Corrective action is only done partially because of findings from external parties.
1	Plan	Policy, planning, program activities are not established.
	Do	Activity is not done.
	Check	Monitoring/review is not carried out.
	Action	Corrective action, not taken.

Based on the parameters of the implementation of optimization of protection and radiation safety on occupational exposure and the fulfillment criteria as described above and refer to **TABLE 3** for scoring criteria then a list of assessment criteria for each parameter is compiled, as shown in **TABLE 4** below. This assessment list is made generic, can be used for diagnostic and interventional radiology or radiotherapy or nuclear medicine facilities.

TABLE 4. Assessment criteria

Parameter	Score	Criteria*)
Management perspective		
Radiation protection and safety organization	5	RPSO is formed officially, documented, integrated into the parent organization's management structure/system, and had an effective performance/program.
	4	RPSO is formed but not officially documented, effective performance/program
	3	RPSO is formed and performance/programs are not monitored and reviewed.
	2	RPSO is formed but only for permit purposes.
	1	RPSO has not yet been formed.
Worker competencies and qualifications	5	Workers' competencies and qualifications are determined and maintained effectively and regularly reviewed
	4	Workers' competencies and qualifications are determined, training programs are implemented, personnel performance reviews have not been conducted
	3	Workers' competencies and qualifications are determined, only part of the training program is implemented
	2	Workers' competencies and qualifications are determined without a training program
	1	Workers' competencies and qualifications have not been determined.
Worker protection against radiation risk	5	Workload design/worker scheduling systems, and workers' dose and health monitoring programs are established, implemented, and evaluated effectively.
	4	A dose monitoring and worker health monitoring program is established, implemented, and evaluated, but does not provide feedback in planning schedules/workloads.
	3	Dose monitoring and health monitoring of workers is carried out but no evaluation is carried out.
	2	A dose monitoring and worker health monitoring program is conducted, but it is inconsistent or irregular.
	1	Workload design/worker scheduling system, dose monitoring program, and worker health monitoring program were not established.
Provision of radiation protection equipment and reliable supporting facilities	5	The procurement and quality assurance program for the completion of radiation protection and its supporting facilities is established, implemented, and controlled and evaluated.
	4	The procurement and quality assurance program for providing radiation protection equipment and supporting facilities is established, implemented, and controlled but not evaluated.
	3	A procurement and quality assurance program for providing radiation protection equipment and supporting facilities is established, implemented but not controlled and not evaluated.

	2	The procurement and quality assurance program for the completion of radiation protection and its supporting facilities is not specified, implementation is only as needed, not controlled, and not evaluated.
	1	The procurement and assurance program for the quality of radiation protection equipment and supporting facilities have not been established.
Implementation of effective communication	5	Communication and information related to worker doses, radiation protection, and safety and safety culture are established and carried out vertically and horizontally at each level of the organization, supported by an effective information system
	4	Communication and information related to worker dose, radiation protection, and safety and safety culture are carried out vertically and horizontally at each level of the organization, supported by information systems, but there is no evaluation/feedback.
	3	Information related to workers' doses, radiation protection and safety, and safety culture is discussed at the meeting as a priority and distributed to each person.
	2	Information related to workers' doses, radiation protection and safety, and safety culture is distributed to each person.
	1	Communication and information related to radiation protection and safety and safety culture are not a priority.
Increasing awareness and involvement of workers	5	The involvement of personnel/workers in planning, monitoring, and evaluation related to the implementation of radiation protection and safety is documented and effectively implemented.
	4	Personnel/workers are always involved in planning, monitoring, and evaluation related to radiation protection and safety, but they are not consistent
	3	Personnel/workers are involved only in monitoring and evaluation related to radiation protection and safety,
	2	Personnel/workers are only involved in planning related to radiation protection and safety.
	1	Personnel/workers only accept policies in planning, monitoring, and evaluation related to radiation protection and safety
Operational perspective		
Dose constraint for radiation workers at the operational stage	5	Dose constraints for radiation workers are established, implemented, documented, and reviewed.
	4	Dose constraints for radiation workers are established, implemented, documented, but not reviewed.
	3	Dose constraints for radiation workers are established, implemented, but not documented and not reviewed.
	2	Dose constraint for radiation workers is established, but not implemented,
	1	Dose constraint for radiation workers are not established
Annual dose monitoring	5	Worker dose monitoring programs are established, implemented, documented, and reviewed
	4	Worker dose monitoring programs are established, implemented, documented but not reviewed
	3	Worker dose monitoring program is established, implemented, but not documented and not reviewed
	2	Worker dose monitoring program is established but is not implemented
	1	Worker dose monitoring program is not established
Quality control or calibration of radiation protection equipment.	5	Quality control or calibration program is established, implemented, evaluated, and documented.
	4	Quality control or calibration program is established, implemented, and documented but not evaluated.
	3	Quality control or calibration program is established, implemented, but not evaluated and not documented.
	2	Quality control or calibration program is established, but not implemented.
	1	Quality control or calibration program is not established
Arrangement of assignment of workers	5	Arrangement of the assignment of workers in the area of radiation according to the type of position and type of work as well as the length of time the work is established, implemented, documented, and evaluated.
	4	Arrangement of the assignment of workers in the radiation area according to the type of position and type of work as well as the length of time the work is established, implemented, documented but not evaluated.
	3	Arrangement of the assignment of workers in the area of radiation according to the type of position and type of work as well as the length of time the work is established, implemented, but not documented and not evaluated.

	2	Arrangement of the assignment of workers in the area of radiation according to the type of position and type of work as well as the length of time the work is set but not implemented.
	1	Arrangement of the assignment of workers in the area of radiation according to the type of position and type of work as well as the length of time the work is not set.
Minimization of dose exposure to workers.	5	Exposure monitoring programs in the work area and optimizing the use of radiation protection equipment during work are established, implemented, documented, evaluated.
	4	Exposure monitoring programs in the work area and optimizing the use of radiation protection equipment during work are established, implemented, documented, but not evaluated.
	3	Exposure monitoring programs in the work area and optimizing the use of radiation protection equipment during work are established, implemented, but not documented, and not evaluated.
	2	Exposure monitoring programs in the work area and optimizing the use of radiation protection equipment during work are established but are not implemented.
	1	Exposure monitoring programs in the work area and optimizing the use of radiation protection equipment during work are not established

*) Details of fulfillment for these criteria can be referred to in the section above

After scoring as in **TABLE 4**, then the quantification of the optimization application is calculated using Equation 1:

$$OP = \left(\left(50\%.\frac{X_a}{Y_a}\right) + \left(50\%.\frac{X_b}{Y_b}\right)\right).\frac{100}{n} \qquad \text{(Equation 1)}$$

with:
OP = total value obtained for the optimization implementation (which has been normalized to a scale of 100)
X_a = the sum of the score in management aspects
X_b = the sum of the score in operational aspects
Y_a = number of parameters in the management aspect
Y_b = number of parameters in operational aspects
n = number of score = 5

If there are any parameters that are not relevant to be applied at the facility, the value becomes N/A and does not count towards the number of parameters or the total score.

After the OP value is known, facility can interpret the performance evaluation through the criteria according to **TABLE 5** below.

TABLE 5. Interpretation of the performance evaluation

OP value	Interpretation
81 - 100	The performance of the facility in optimizing radiation protection on occupational exposure is very good
61 - 80	The performance of the facility in optimizing radiation protection on occupational exposure is good
41 - 60	The performance of the facility in optimizing radiation protection on occupational exposure is moderate
20 - 40	The performance of the facility in optimizing radiation protection on occupational exposure is poor

The assessment should be carried out with a period of at least 1 year and followed by a review/analysis. The results of the assessment must be stated in the form of a report which includes details of the results, review/analysis of results, and recommendations so that it is easy to design an appropriate follow-up program.

Analysis of Results and Follow-Up

The result obtained must be analyzed to see an overview of each parameter. The analysis should not only focus on the quantification of the total value but also observe the assessment result for each parameter so that it can be identified parts or aspects that need improvement.

The result obtained must also be compared with previous years and analyzed to see a picture of trends from year to year. The trend of the results must be visualized graphically so that the following 3 (three) possible situations will occur:

a. Tends to increase, which means that efforts to optimize radiation protection have been carried out effectively.

b. Tends to be constant, which means that radiation protection of workers have been carried out optimally, but have not been effective because they do not show improvement.
c. Tends to decrease, which means that efforts to optimize radiation protection are not effective.

Situations such as in points b and c certainly require follow-up i.e the facility must conduct a thorough review to find the root causes and take appropriate corrective actions. Corrective action is an action to eliminate the root cause of a non-conformity to prevent the discrepancy from recurring [15].

If no comparison is made with the results from the previous year (because it is the first time done) then it can be analyzed on the parameters that indicate deficiencies.

CONCLUSION

The assessment parameters of the implementation of radiation protection optimization on the occupational exposure can be approached from a management perspective and an operational technical perspective.

The method proposed in quantifying performance assessments based on the implementation optimization of radiation protection on occupational exposure is a rating scale with the criteria for each score adopting the fulfillment of the PDCA principle.

Assessment can be a tool to monitor the effectiveness of the implementation of radiation protection optimization on occupational exposure so that facilities will be easy to improve radiation protection and safety systems for radiation workers continually, and BAPETEN will be easy to carry out monitoring and assessment.

ACKNOWLEDGMENT

The study was financed by DIPA BAPETEN 2018 and carried out in cooperation with P2STPFRZR team for collecting data, respondents for joining the survey, and experts who provide technical advice.

REFERENCES

[1] BAPETEN, *Peraturan Kepala BAPETEN Nomor 1 Tahun 2017 tentang Pelaksanaan Inspeksi dalam Pengawasan Pemanfaatan Tenaga Nuklir*. Indonesia, 2017.
[2] E. Kunarsih, "Penguatan Perlindungan Pekerja Radiasi di Fasilitas Kesehatan Melalui Self Assessment Efektivitas Optimisasi Proteksi Radiasi: Sebuah Tinjauan," *Pros. Semin. Keselam. Nukl. 2019*, 2019.
[3] IAEA, *IAEA Safety Glossary: 2018 Edition*, 2018th ed. Vienna: IAEA, 2019.
[4] *Peraturan Pemerintah Nomor 33 Tahun 2007 tentang Keselamatan Radiasi Pengion dan Keamanan Sumber Radioaktif*. Indonesia, 2007.
[5] Bapeten, *Peraturan Kepala BAPETEN Nomor 4 Tahun 2013 tentang Proteksi dan Keselamatan Radiasi dalam Pemanfaatan Tenaga Nuklir*. Indonesia, 2013.
[6] E. Kunarsih, *Pedoman Teknis Penetapan dan Penerapan Pembatas Dosis Pekerja Radiasi di Fasilitas Kesehatan*. Jakarta: P2STPFRZR, BAPETEN, 2017.
[7] E. Kunarsih, *Pedoman Teknis Pengawasan Penerapan Optimisasi Proteksi dan Keselamatan Radiasi di Fasilitas Kesehatan*. Jakarta: P2STPFRZR, BAPETEN, 2018.
[8] IAEA, *General Safety Requirements Part 3 No. GSR Part 3: Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards*. Vienna: IAEA, 2014.
[9] E. Kunarsih and Iswandarini, "Skenario penetapan dan penerapan pembatas dosis di fasilitas kesehatan dalam upaya optimisasi proteksi radiasi tahap operasional," in *Seminar Keselamatan Nuklir 2018*, 2018.
[10] IAEA, *Safety Reports Series No. 21: Optimization of Radiation Protection in the Control of Occupational Exposure*, vol. 147. Vienna: IAEA, 2002.
[11] Pusat Bahasa Depdiknas, *Kamus Bahasa Indonesia*. Jakarta: Departemen Penidikan Nasional, 2008.
[12] ISO, *ISO 9004 Managing for the Sustained Success of an Organization - A Quality Management Approach*. Switzerland: ISO, 2009.
[13] IAEA, *General Safety Guide No. GSG-7: Occupational Radiation Protection*. Vienna: IAEA, 2018.
[14] M. Singaribun and M. Effendi, *Metode Penelitian Survey*. Jakarta: LP3ES, 1989.
[15] ISO, *ISO 9000:2015 Quality management systems - Fundamentals and vocabulary*, 4th ed. Switzerland: ISO, 2015

The Challenges of Structures, Systems, and Components (SSC) Classification on Next Generation Reactor

Arifin Muhammad Susanto[1, a)]

[1]*Center for Assessment of Nuclear Installation and Material, BAPETEN.*

[a)] a.msusanto@bapeten.go.id

Abstract. Challenges arise when involved in innovation and technology on developing nuclear installation. A new type of reactor on generation IV come up with remarkable ideas, innovations, and methods, as well as the use of materials that no one ever heard before. Technology vendors are competing to show the world how advantage, simplicity, and integration of their reactor and which could eliminate all the problems we have seen years on typical light water reactor types such as LOCA, CDF, etc. Dealing with the first of a kind type of reactor leads to technology, manufacture, and licensing and regulation framework readiness and lack of operational and expertise experience. The idea of an integrated reactor could reduce the licensing and manufacturing time frame somehow not necessarily true, the fact that some material requires high heat strength requirements during the operational stage for instance. The manufacturing process of those materials also based under SSC's classification pre-defined involves a thorough set of tests to get approval by the regulator, as well as premature code and standard available. This paper was based on observational research which gives an idea to identify difficulties in developing SSC classification that highlighted a new type of reactor. Things to be considered and realized what the challenge to be faced in the future. However, from the challenges identified, the potential solutions which could resolve are from research development continuity, legal framework harmonization, and development of code and standards.

Keywords: SSC, Next Generation Reactor, Classification

INTRODUCTION

Nuclear safety can also be defined as the protection of humans and the environment against hazards arising from damage, failure of the nuclear system, or/by human error. Nuclear safety is related to the risk of radiation in normal circumstances and also the risk of radiation as a consequence of an incident and other direct consequence that may occur due to loss of control on the core of a nuclear reactor, radioactive source, or other radiation sources. Safety measures include preventive measures, as well as arrangements, that have to be taken to reduce the consequences that occurred.

Safety is achieved by the effective defense against radiation hazards at nuclear installations. Nuclear plant safety measures include technical and management safety. Safety techniques include site monitoring, design and construction, commissioning, operation, modification, decommissioning, and verification and safety assessment.

Safety functions are specific functions that must be fulfilled to achieve the safety of a facility or activity to prevent radiological consequences in normal operation, anticipated operating occurrences, and accident conditions. The power reactor safety system is an important safety system that is provided to ensure a safe shutdown of the reactor and the disposal of residual heat. Safety systems are needed to limit the consequences of anticipated operational occurrences and design accidents. The fundamental safety functions for nuclear power reactors include reactivity control, cooling of radioactive material, and confinement of radioactive material. Radioactive material cooling functions include heat disposal from reactors and fuel storage. Radioactive material confinement functions include radiation shielding and planned radioactive release control as well as restrictions on radioactive release in the event of an accident.

International Atomic Energy Agency (IAEA) SSR 2/ rev.1 stated that systems, structures, and components (SSC) are terminology for expressing all elements in a facility that contribute to protection and safety functions. The components themselves are interpreted as discrete elements of a system, such as motors, pumps, and tanks. The structure is defined as a passive element, for example, buildings, vessels, and shields. A system is a number of components arranged in such a way as to form a certain function [1].

According to IAEA SSG 30, it is very important for installation to have SSCs that are capable of performing safety functions. This is necessary so that the design of the plant meets the safety requirements [2].

The project design (both conceptual and basic design) begins with several processes related to safety and engineering processes, namely method definition, the safety features, combined SSC, and general installation layout. The design of the factory was later developed by many engineering disciplines: such as mechanical engineering, electrical (electronic and power), civil structural, instrumentation and control system, and other subject matter. The interaction and interface between all engineering disciplines are very important throughout the project engineering life cycle to make sure comprehensive consideration of functional requirements. It will eventually reduce or eliminate egocentric which to become silo effects.

To control the reactivity using the control rod is an example of a very typical on pressurized water reactor (PWR). The designer gives two modes of a control rod operation those are: a normal mode and an 'emergency' mode. Since performs very crucial action, hence control rod was defined as a safety system.

The next-generation reactor also requires SSC to perform its function, regardless of safety or non-safety for instance pump, water & drain tank, and soon. The above example of a control rod could not be applied, but in function, the safety system reactor must be performed satisfactorily. Safety systems are the parts needed to ensure the safety of plant operations. Safety systems in nuclear reactors are necessary to ensure the shutdown of the reactor safely, disposal of residual heat from the reactor inner core, or to minimize and mitigate the radiological consequences of anticipated operational events (AOO), and design basis accidents (DBA).

Up to now, code and standards available mostly based on pressurized water reactor (PWR) and boiling water reactor (BWR) types. By continuous improvement, the code and standard are well established. The experiences and technical skills have been developing significantly. Meanwhile, in contrast with the lack of experience in dealing with the next-generation reactor, for example on regulating new material, new safety concepts, etc.

At some point, even its already established, there have been identified some mismatch or a discrepancy among code and standards for PWR & BWR. The next-gen reactor gives an idea of applying advanced technology, greener material selection, which somehow traps in a gap between innovation and code and standards, and regulatory frameworks. Especially for embarking countries which been offered a new type of reactor.

This paper gives an idea to identify difficulties in developing SSC classification highlighted a new type of reactor in reflect current regulatory frameworks. Things to be considered and realized what the challenge to be faced in the future from research development continuity, legal framework harmonization, and development of code and standards.

Regulatory Envelopes

Requirement No. 22 on IAEA No. SSR-2/1 [1] mentioned: All items (SSC) important for safety have perform identification and classification according to their function of safety and significance. Requirement 30 of SSR-2/1 (Rev. 1) [1] states: “A qualification program for items important to safety shall be implemented to verify that items important to safety at a nuclear power plant are capable of performing their intended functions when necessary, and in the prevailing environmental conditions, throughout their design life, with due account taken of plant conditions during maintenance and testing”.

Safety requirements that must be met from a nuclear reactor design include:

- The ability to safely shut down and maintains safe shutdown conditions during and after operating conditions and accident conditions.
- Ability to dispose of residual heat from the reactor core, reactor and fuel in storage after the reactor has been shut down and during and after operational conditions, and accident conditions.
- Ability to reduce the potential for radioactive material release and ensure that each release is within the required limits during and after operating conditions and within acceptable limits during and after the design accident.

IAEA SSR-2/1 [1] also specifies the following main safety classification criteria:

- 5.34. Classification of SSC importance to safety is conducted based on deterministic methodology and supported by probabilistic methodology if possible, factors to be taken as follows:

    1. The function of safety from SSC to be achieved;
    2. The failure consequences if the function of safety fail to achieve;
    3. The rate of frequency of the SSC initiated to achieve function of safety;
    4. The time or period of the accident events (PIE) that the SSC is initiated to achieve its safety function.

- 5.35. By design, no interference between SSC importance to safety is occurred as well as there is no failure due to different classes of safety.
- 5.36. The multiple function which been achieved by SSC, hence the SSC is classified as safety class.

The IAEA SSG-30, Structural, System, and Component Safety Classification at nuclear power plants [2] mention how to comply with the requirements on safety classification of SSR 2/1 as described above. The overall methodology is to provide SSC important to safety identification and classification referring to their function as well as signs of safety. According to the SSG-30, the safety class of SSC will ultimately protect workers and people, and the environment.

Based on most of the criteria specified by SSR-2/1, the IAEA SSG-30 define SSC class of safety into three classes (1, 2, and 3) and explains the basics of classification.

Regulations on SSC classification is also found in the United States Nuclear Regulatory Commission (USNRC). Those divide the classification into two categories: 1) safety-related or 2) unrelated safety. According to 10 CFR §50.2 [3] describes the safety-related SSC as the SSC which performs to achieve its function to maintain the integrity of limit pressure of cooling on the reactor, to perform safe shutdown, and to perform prevention and if already occurred then to perform mitigating of the accident consequences.

Other NRC requirements also being apply, for example SSC which perform as fire protection, to mitigate events of anticipated transients without scram (ATWS) and station black out (SBO). Those SSC requirements are determined based on plant design and operation. Those USNRC regulation are intended for LWR type reactor.

On the other hand, the legal basis for national regulations related to the safety classification process by Indonesian Regulation is as following:

- Government Regulation number 54 of 2012 concerning Nuclear Installation Safety and Security [4] article 13, paragraph 1: "To meet the general requirements and special design requirements as referred to in Article 10 paragraph (2), permit holders must determine the classification of structures, systems, and components. nuclear installation ", paragraph 2:" The classification referred to in paragraph (1) shall be carried out based on the safety class, quality class, and/or seismic class. "
- BAPETEN Chairman Regulation number 3 of 2011 concerning Safety Provisions for Power Reactor Design [5], Article 34, paragraph 1: "Permit holders must identify and classify structures, systems, and components, including software for instrumentation and control, based on their importance to the safety function", and from Article 37 "The structures, systems, and components referred to in Article 34 must be classified based on quality and seismic class".

Our in-house BAPETEN Agency Regulation for guiding classification of SSC nuclear installation is currently undergoing legal harmonization between the Ministry of Legal and Human Rights and BAPETEN, and expected to be published tentatively on next year. This document was developed based on the IAEA safety series document.

According to nuclear reactor evolution, currently, we are at the stage of generation III type reactor. Generation I reactor was built at around the 1950's such as Magnox, Fermi I, followed Gen II (1980's) reactors of CANDU, early PWR and BWR, and Gen III/+ (1990-now) such as APR1400, and ABWR. The Gen III intends to get standardize of reactor types, increase more on safety by developing design robustness, lifetime operational to get more longer without compensating on operational cost, and more economic [12]. The next generation of reactor (Gen IV) development program was proposed by several states member in 2002 during Gen IV International Forum (GIF). They were proposing a long term (and/ joint) research to develop several types of reactor that might be proposing for near future. Some of which are Molten salt reactor (MSR), High Temperature Reactor (HTR), and Sodium-cooled reactor [13].

An advance nuclear reactor is described as a nuclear power plant which has significantly improved over the last type reactor. A gen IV reactor usually refer to advance reactor, since the new-gen propose high level of safety by promoting inherent safety features, efficiency on operational, reduce risk of security, advance technology related to material technology and modular type [14].

GENERAL CLASSIFICATION PROCESS

The classification process may involve long and pain taking processes for all entities, such as vendors, manufacturers, and regulators. Here is some flow process gathered from two regulations.

IAEA TECDOC 1787 on Application of the Safety Classification of Structures, Systems, and Components in Nuclear Power Plants provide to assist any organization in establishing a comprehensive safety classification of SSCs compliant with the IAEA recommendations and to capture all SSCs to be classified and to assign each of them to the appropriate safety class to reflect its own safety significance [6]. The document method uses two approaches: function and design provision identification. Each approach follows different steps include verification of results.

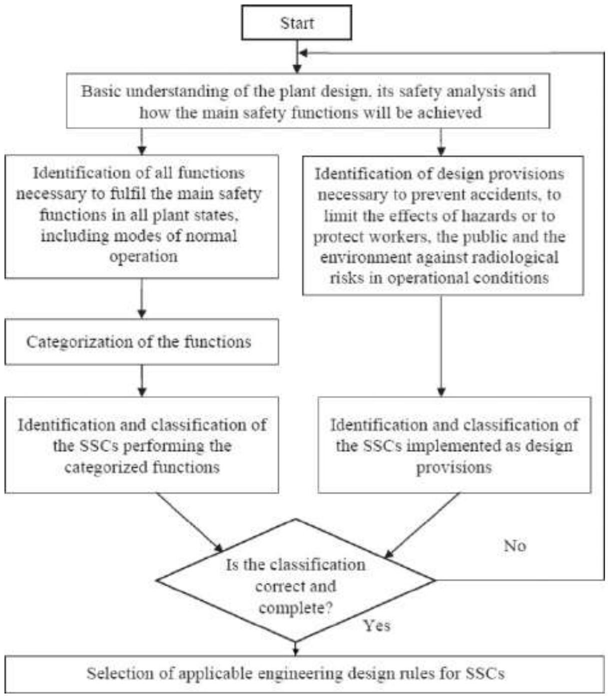


FIGURE 1. Classification process flowchart [2][6].

In general, two approaches on SSC classification were applied, both are identifying the function as well as identifying the design provision on related SSC against the effect of hazards, and others. Both methods come up with SSC identification, hence two approaches were verified to get the most suitable classification SSC which fit in the requirements, iterative process was applied if found not suitable.

Before establishing the classification, the plant states to be considered within the design are defined under IAEA SSR-2/1 [1] and include accident conditions with core melt. Plant states are usually defined as follows:

Plant states considered in the design				
Operational states		**Accident conditions**		
Normal operation	Anticipated operational occurrences (AOO)	Design basis accident (DBAs)	Design Extension conditions (DEC)	
			Without significant fuel degradation	With core melting

FIGURE 2. Plant states in designed [1].

That IAEA document is guiding for NPP regardless of the type of reactor, but it can be assumed from the period of publishing, only cover light water reactor e.g. pressurized water reactor (PWR) and boiling water reactor (BWR) which mainly generation III/III+ type reactor.

Idaho National Laboratory (INL) has produced several documents, highlighted on the future development of high next-generation nuclear power plant. The documents produced under the Next Generation Nuclear Plant (NGNP) Project Research & Development. Those documents tend to identify the applicability of a potential new type of reactor to be built and ultimately to support the commercialization of high-temperature gas-cooled reactor (HTGR) technology.

We realized that not all elements of current regulations (and their related implementation guidance) apply to HTGR technology or the next-gen reactor at nowadays. Certain policies established during past LWR licensing actions must be realigned or modified to properly accommodate advanced technology.

The NRC's policy on Advanced Regulation of Reactors [7] is to ensure adequate protection of the environment and public health and safety. For the newly, the advance reactor will have to at least an innovation feature and enhancement on increasing safety margin, applying inherent safety, and emphasizing a passive safety system.

As mentioned, NRC divides the class into safety-related and non-safety-related. For non-safety-related SSC, NRC uses further consideration to apply special treatments that reflect their significance of safety.

The article of INL/EXT-10-19509 - Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper [8] provides general guidance to defines the methods to classify safety on SSC.

The classification is designed for a new reactor to refer to high-temperature gas-cooled reactor technology (HTGR).

From the same document, NRC was proposing SSC identification for this type of reactor. They divide into safety and non-safety-related. For SSC on safety-related has to perform safety shutdown and also mitigate include prevent the accident consequences. Whereas for non-safety-related specifically with special treatment, the SSC must perform safety function for mitigating AOO, as well as preventing the DBE frequency, as depicted in the figure below.

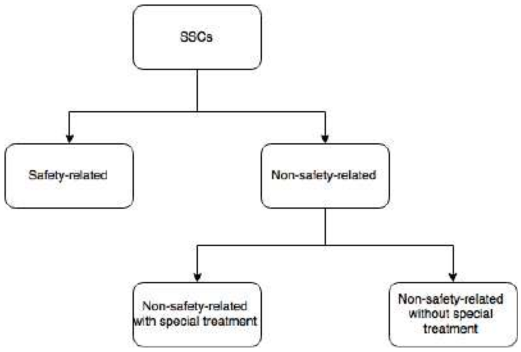


FIGURE 3. SSC Classification according to NRC [8].

Firstly, selection process license bases event (LBE) and the second determination of the top-level regulatory criteria (TLRC). TLRC will put a set of value on consequences frequency, and radiological consequences. Hence those values are going to be used as a value on classification and evaluating the LBS [9]. LBE is a selection of events that are considered during the licensing process, those event/s are used to define the requirements [15]. According to the INL/EXT-10-19509 [8], the process begins with identification on generic TLRC, a set of values of acceptable consequences or risks determined from NRC regulations. During the process, three regions of frequency are chosen: Anticipated Operational Occurrences (AOOs)a Design Basis Events (DBE), and Beyond DBE (BDBEs).

The process of SSC classification starts with specifying the safety function for the DBE's of the respective SSC. Those SSCs during the events of DBE must or keep available and sufficiently capable, and reliable to perform it's the safety function.

Generic examples regarding SSCs providing functions leading to SSC safety classification as safety-related, non-safety-related with special treatment, and non-safety-related, are shown in Figure below.

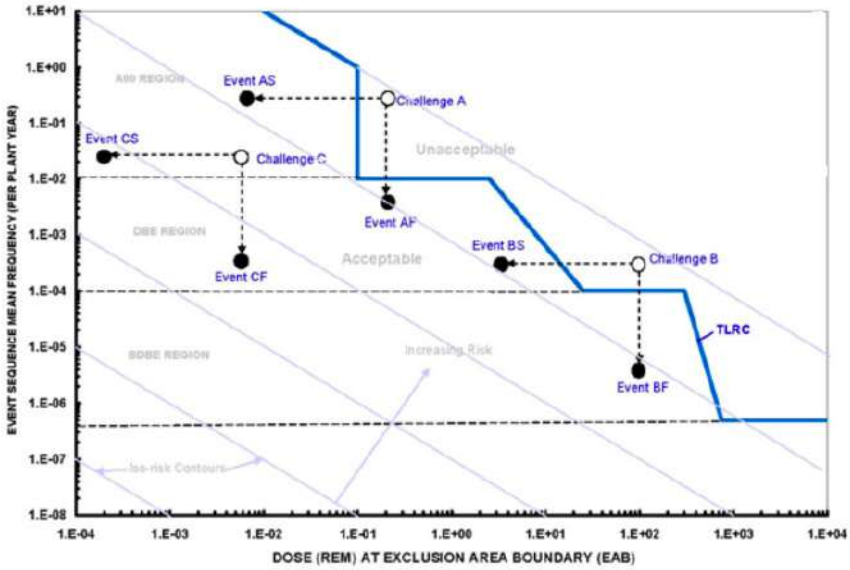


FIGURE 4. Impact of safety classified SSCs in prevention and mitigation of LBEs. Figure taken from [8].

From the figure above showing how the SSC were classified, based on events of likelihood. Two challenges were determined. Classification of SSC was selected on how they respond to mitigating the result of occurrences. SSC AS is an example of an SSC whose successful performance mitigates (ensures) the results of an occurrence

AS and remains below the TLRC thus preventing AOO (Challenge A); SSC AF is an example of an SSC whose successful performance prevents the frequency of a corresponding DBE (Event AF), whose consequences exceed the AOO dose criteria, from getting into the AOO region. The SSC will be classified as NSRST.

The special treatment applied to the SSC category currently helps to organize the appropriate LBE (AS and AF) within the respective LBE category.

Two methodologies gathered from different regulation shows different approached, not mentioned other code and standards which provides different methods. The challenge is which methods are suitable for classifying the SSC next-gen reactor. However, the later document was proposed for HTGR type but no reactor for this type built yet.

Table 1 below shows the different safety classification schemes used by the main international standards organizations.

CLASSIFICATION CHALLANGES

As a glimpse already presented, showing that classification of SSC in my opinion facing a big challenge rather than continuing what has been established during the operation of NPP in decades. Some challenges are identified as described below.

<u>Unclear requirements for safety classification.</u>

Codes and standards are established based on their consensus. Some requirements could have missed interpretation due to not identified. An ambiguous requirement will lead to a different interpretation. Vendors, utility, and authority understand and considering differently, which will impact to potential delay or postponed of the project schedule. Low-level regulation tends to present quantification or parameter to describe requirements and acceptance criteria. Each of low-level regulation somehow showing an overlap criterion. For instance, requirements produced by the regulator put acceptance criteria more conservative rather than from industry. The following keywords frequently cause trouble in the interpretation of requirements [10], for example on instrumentation and control (I&C) systems:

- Defense-in-depth and diversity (assignment of different I&C systems and provision of diversity within and between systems to reduce the likelihood that common cause failures within the I&C system will cause failure of reactor safety functions).
- Separation (physical separation/ electrical isolation/functional independence/independence of communication).
- Redundancy (level of required redundancy e.g. N+1/N+2).
- Reliability/availability (limits for digital I&C systems).
- False activation (inadvertent actuation of I&C functions).

The wording (on **TABLE 1**) has a slightly different definition. Some word probably has the same meaning but the level of depth, criteria are somewhat different.

<u>The difference in International and National Regulations.</u>

In an industrial country as well as who has operation of NPP, usually industrial and utility association has already established. They are joining together in a consortium to develop a common understanding among them on how to operate their system safely and effectively. Industrial or nuclear utility association produced in-house regulation mandatory for members only, e.g. WENRA, ICE, IEEE, STUK, etc. That resulted in inconsistency between international (IAEA) and national standards (STUK).

Manufactural and nuclear industry set a different level of standard criteria. Even between International standards present challenges to equally level the criteria. An example taken from the World Nuclear Association [10] about The IAEA SSG-30 sets the function for the main installation parameter control (MPP) to Class 3. Meanwhile, another code IEC 61226 put it to Class 2. It will misguide applicants or users. Agreement on which code and standard is going to apply is very crucial

<u>Lack of experience.</u>

The next-gen reactor will be proposed to build for embarking countries or others. The country has a city or industrial district quite far for the main electricity grid. The next-gen reactor produces smaller electricity power capacity rather than a large NPP. The capacity of the reactor is in the range of 10 to 250 MWe. Hence, it is suitable for low power electricity demand, smaller grid infrastructure, and on top of that, flexibility on transporting the SSCs.

For the first of a kind reactor, it’s common practice to use a turn-key project scheme for the first reactor development. The vendor will provide all requirements such as service, technology, and, depending on the contract agreement, include reactor operational and maintenance staff and service. However, if another contract scheme that requires local industrial participation is selected than various resources need to be prepared. Regulation has to be developed to envelope such kind of consideration, as well as other codes and standards to assist the regulation. Experience and scientific knowledge of reactor technology, material, operational subject to further enhancement. Especially on safety analysis, which is required to assess risks and consequences, so later, review of the classification of SSC resulted in a comprehensive report.

TABLE 1. Gap analysis International standard on safety system classification [10]

Organizations or Countries		Safety Classification of I&C Functions and Systems in nuclear plants			
Main international standard organizations					
IAEA NS-G-1.3		Systems Important to Safety			Systems not important to safety
		Safety	Safety-related		
IAEA SSG-30	Function	Safety category 1	Safety category 2	Safety category 3	
	System	Safety class 1	Safety class 2	Safety class 3	
IEC 61226		Systems Important to Safety			Non-classified
	I&C function	Category A	Category B	Category C	
	I&C system	Class 1	Class 2	Class 3	
IEEE		Systems Important to Safety			Non-safety-related
		Safety-related	[5]		
EUR[6]	Safety level of functions / I&C systems	F1A	F1B	F2	NS (non-safety)
MDEP member states					
Canada		Category 1	Category 2	Category 3	Category 4
France		F1A	F1B	F2	Non-classified
Finland		Class 2	Class 3	EYT/ STUK	EYT (classified non-nuclear)
UK		Class 1	Class 2	Class 3	Non-classified
United States		Systems Important to Safety			(not specified)
		Safety-Related	[5]		
India		IA	IB	IC	NINS
Japan		PS1/MS1	PS2/MS2	PS3/MS3	Non-nuclear safety
Korea		IC-1	IC-2		IC-3
Russia		Class 2	Class 3		Class 4 (Systems not important to safety)
Others nuclear states					
Switzerland		1	2	3	Non-classified
Germany	I&C function	Category 1	Category 2	Category 3	Non-classified
	I&C equipment	E1		E2	

IAEA terminology on graded approach or so-called risk-informed classification of SSC on NRC regulation will play a major role to identify and classify the proposed nuclear power plant. It's obvious, technology vendor of NGNP will treat their installation majority as industrial grade. Due to the claimed that it will pose less or even to have zero risk of a radiological release. Pebble bed fuel coated particles stable to beyond maximum accident temperatures. Its ceramic fuel retains radioactive materials up to and above 1800˚C. Thus, no large early release of fission products from the fuel [11]. Generation IV reactor safety features inherently reduce the risk probability of events.

However, the safety perspective adheres to the new modern aphorism, "there is no such thing as zero risks, only acceptable risk".

Ongoing R&D on Material Science

The next-generator reactor mostly incorporated with a high-temperature environment, fast neutron reaction, and also utilized cogeneration for other purposes. Those 'extreme' conditions and environments require high reliability and stability material to work under circumstances.

Any material deficiency and failure such as fatigue, crack, swelling, and growth as well as aging on mechanical properties include tensile strength, creep, resistance, corrosion, etc., that need to be solved.

To comply with those requirements, research, and development (R&D) has to be done by assessing and qualifying any commercial material available.

Optimization and (if needed) new development of material are required to work against such irradiation, high heat, stress, high neutron flux, and corrosion conditions. On top of that, the material has to serve for the operational lifetime of the reactor to keep reactor safety uncompromised.

From the ongoing R&D, several materials have potentially been promising for next-generation reactors. Those candidates would be Ferritic/martensitic steel, oxide dispersion strengthened steel, ceramic, and nickel-based alloy [16]. Those selections are taken from modeling analysis, further experiment is required.

SUMMARY

Generation IV reactor or Next-generation reactor uses the simplification of large SSC. The integration of SSCs into one system is proposed to reduce the potential failure of the systems. Integration of the system to perform several functions in one SSC will impact material to expose operational environment conditions, e.g. high temperature and pressure.

The development of material selection for the next-generation reactor incorporates advanced technology. R&D on the material is still ongoing. In the current condition, there are still some barriers to be break if the new reactor being operated. A breakthrough in material developments is needed.

The selection of code and standard for classification of SSC needs further consideration, to prevent issues in the future. The issues arise such delay on the licensing process, difficulty in material manufacturing which leads to postponing on the project schedule.

Code and standard harmonization, probably one of the solutions to solve the issue, means to have a common understanding to interpret requirements and criteria, but to use one well-known standard is the ideal way to classify the SSC.

The robustness of legal infrastructure, as well as the test and qualification process, accounted for is a means to fulfill the safety of the system satisfactorily.

REFERENCES

[1] IAEA Specific Safety Requirements (SSR) 2/1 (Rev.1), Safety of Nuclear Power Plants: Design, Vienna, 2016

[2] IAEA Specific Safety Guide (SSG) 30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, Vienna, 2014

[3] US NRC Title 10 Code Federal Regulation (CFR) §50.2, Definitions

[4] Government Regulation number 54 year 2012 concerning Nuclear Installation Safety and Security

[5] BAPETEN Chairman Regulation number 3 of 2011 concerning Safety Provisions for Power Reactor Design

[6] IAEA TECDOC 1787 on Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants.

[7] NRC-2008-0237, United States Nuclear Regulatory Commission, Policy Statement on the Regulation of Advanced Reactors.

[8] INL/EXT-10-19509 - Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper

[9] INUEXT-10-19521 - Next Generation Nuclear Plant Licensing Basis Event Selection White Paper

[10] Safety Classification for I&C Systems in Nuclear Power Plants - Current Status & Difficulties, CORDEL Digital Instrumentation & Control, World Nuclear Association Published: September 2015, Report No. 2015/008

[11] Reitsma, F., Introduction to the CRP on modular HTGR Safety Design: Background and Overview; Definition of modular HTGR, Status, Joint IAEA–GIF Technical Meeting on the Safety of High Temperature Gas Cooled Reactors (EVT1703158)

[12] Farkas, G., From Gen I to Gen III, Presentation file on IAEA meeting, 2010

[13] ANS-Center for Nuclear Science and Technology Information, The Next Generation of Reactors, http://nuclearconnect.org/know-nuclear/technology/the-next-generation-of-reactors, taken Sept 03, 2020, at 09:41 AM Jakarta.

[14] Congressional Research Service, Advanced Nuclear Reactor: Technology Overview and Current Issues, CRS Report, April 18, 2019

[15] US Design Certification, Licensing Basis Event Selection for The Pebble Bed Modular Reactor, Document Number 040251, 2006.

[16] Yvon, P., Carre, F., Structural materials challenges for advanced reactor systems, Journal of Nuclear Materials 385 page 217-222, Elsevier, 2008

Assessment of Safety Provision on Design of the Reactor Cooling System and Associated Systems (RCSAS) in Nuclear Power Plants: Focused on Design General Requirement

Arifin Muhammad Susanto[1, a)] and Sihana[2)], Nur Siwhan[3)]

[1]*Center for Assessment of Nuclear Installations and Material, BAPETEN*
[2] *Department of Nuclear Engineering and Engineering Physics, UGM*
[3]*Center for Assessment of Nuclear Installations and Material, BAPETEN*

[a)] a.msusanto@bapeten.go.id

Abstract. One of among safety fundamentals in a nuclear reactor is to maintain a sufficient flow of cooling to ensure compliance with fuel design limits. It means it is necessary to identify and assess criteria, parameter design of the reactor cooling system (RCS), and associated systems which to maintain reactor safety. In this paper, a review has been carried out not only cover on the reactor cooling system for light water reactor (LWR) but also in touch on the future most anticipated small modular reactor. Those are high-temperature gas reactor (HTGR) as well as molten salt reactor (MSR). Safety provision on the design of the reactor cooling for three types of the reactor was introduced according to the typical characteristic of its system. This paper also showed recommendations to be taken for following regulation from the draft International Atomic Energy Agency (IAEA) document. This paper is to give recommendations to develop safety provisions on RCSAS regulation by the regulatory body, hence the regulation is useful to provide recommendations for designers and owners who will apply for a license to build and operate a nuclear reactor in the next term.

Keywords: RCSAS, safety provision, design, NPP

INTRODUCTION

Nuclear safety can also be defined as the protection of humans and the environment against hazards arising from damage, failure of the nuclear system, or/by human error. Nuclear safety is related to the risk of radiation in normal circumstances and also the risk of radiation as a consequence of an incident and other direct consequences that may occur due to loss of control on the core of a nuclear reactor, radioactive source, or other radiation sources. Safety measures include preventive measures, as well as arrangements, that have to be taken to reduce the consequences if occurred.

Safety is achieved by the effective defense against radiation dangers at nuclear installations. Nuclear plant safety measures include technical and management safety. Safety techniques include site monitoring, design and construction, commissioning, operation, modification, decommissioning, verification, and safety assessment.

Safety functions are specific functions that must be fulfilled to achieve the safety of a facility or activity to prevent radiological consequences in normal operation, anticipated operating occurrences, and accident conditions. The power reactor safety system is an important safety system that is provided to ensure a safe shutdown of the reactor and the disposal of residual heat. Safety systems are needed to limit the consequences of anticipated operational occurrences and design basis accidents. The fundamental safety functions for nuclear power reactors include reactivity control, cooling the radioactive fuel, and confinement of radioactive material. The cooling function includes heat dissipation from the reactor and fuel storage. The confinement function of radioactive fuel includes shielding for radiation and control of planned radioactive releases also restrictions on radioactive release in the event of an accident. Safety requirements that must be met from a nuclear reactor design include:

- The ability to safely shut down and maintains safe shutdown conditions during and after operating conditions and accident conditions.
- Ability to dispose of residual heat from the reactor core, reactor, and fuel in storage after the reactor has been shut down and during and after operational conditions and accident conditions.

- Ability to reduce the potential for radioactive material release and ensure that each release is within the required limits during and after operating conditions and within acceptable limits during and after the design accident.

The current state, there is no regulation to deal with design provision on RCSAS if applicants or designer decides to develop systems or components of RCSCAS. Normally, they refer to international code and standard which is a best practice in the nuclear industry. However, this paper was intending to assess and identify the following requirement of the reactor cooling system for many types of reactors. Various types of reactors were considering in this study as one step ahead of preparation if the applicant comes up with remarkable ideas. The type of reactor discussed are MSR, HTR, and PWR, hence ultimately, the safety provisions need to be established to provide recommendations for designers and owners who will apply for a license to build and operate a nuclear reactor

Regulatory Envelopes

According to Government Regulation (GR) number 54/2012 and BAPETEN Chairman Regulation (BCR) number 3/2011 on Design Specific Requirement for Nuclear Power Plant (NPP). It's stated that there is three significant safety functions for NPP should be established, those are reactivity control, radioactive fuel cooling, and confinement of radioactive source. Below describes of each regulation deal with the topics respectively.

For reactivity control, here are detail article mentioned about it for both regulations:

- NPP Design Specific Requirement (GR 54/2012),
  - Article 12, point 2, e.g.:
    - Reactor core;
    - Shutdown system;
    - Reactor protection system;
    - Instrumentation & control system.
- NPP Design Specific Requirement (BCR 03/2011)
  - Article 66-70 on Reactor core and associated system;
  - Article 71 on Reactor shutdown system;
  - Article 73-80 on Reactor protection system;
  - Article 81 on the reactor cooling system and associated system.

For radioactive fuel cooling, here is a detailed article mentioned about it for both regulations:

- NPP Design Specific Requirement (GR 54/2012),
  - Article 12, point 2, e.g.:
    - Engineered safety features;
    - Nuclear material storage and handling;
    - Auxiliary system.
- NPP Design Specific Requirement (BCR 03/2011)
  - Article 81-88 on Reactor cooling system and associated system;
  - Article 89 on Emergency core cooling system, and engineered safety features.

Lastly, for Confinement of radioactive source, here is a detailed article mentioned about it for both regulations:

- NPP Design Specific Requirement (GR 54/2012):
  - Article 12, point 2, e.g.:
    - Confinement system
    - Radioactive waste management system
- NPP Design Specific Requirement (BCR 03/2011):
  - Article 90-97 on System and Containment Structure;
  - Article 110 on Radioactive waste management system.

From the description above, the reactor cooling system and its associates play a vital role to ensure the safety of the operation of the Nuclear Power Plant (NPP). The three safety functions are elaborated into 10 specific requirements for nuclear reactor design in PP 54/2012 namely reactor core, heat transfer system, blackout system, reactor protection system, engineered safety features, containment systems, instrumentation and control systems, fuel handling and storage systems, systems of radioactive waste management and auxiliary systems. Furthermore, specific requirements for the design of nuclear reactors are broken down into 13 items in BAPETEN Chairman Regulation (BCR) 3/2011. This paper will systematically describe the design requirements in general for each type of reactor.

NUCLEAR REACTOR COOLING SYSTEM AND IT'S ASSOCIATED SYSTEMS

The reactor cooling system and its associated systems consist of several subsystems including reactor cooling system, connected system, associated system, and ultimate heat sink. The reactor cooling system could be defined as a system consisting of components needed to ensure proper cooling flow (not beyond design limits). Not included in the RCS are the fuel element and the reactivity control element in it. RCSAS describes on PWR type mostly consists of pressure vessels with top cover devices, internal reactor vessels (other than fuel assembly and core support structures), steam generators, reactor cooling pumps, pipelines connecting steam generators, reactor cooling pumps (hot leg, cold leg and steam generator connecting lines and pumps on each line), pressurizer with a release valve and safety valve. The internal pressure vessel also including a control rod directing housing. Meanwhile, the RCSAS for Boiled Water Reactor (BWR) type reactor mainly consists of pressure vessels include an upper device, internal pressure vessels (other than fuel devices, and core support structures) that maintain the flow of primary cooling such as pumps, internal recirculation pumps or separators. Internal pressure vessels include venturi flow, orifices, and control rod directing containers. Steam and feed water pipelines, cooling recirculation systems such as pumps, pipes, and valves are also part of the reactor cooling system.

The reactor cooling system for Molten Salt Reactor (MSR) type generally includes reactor vessels, fuel lines in the reactor core, fuel circulation pump, the primary part of the intermediate heat exchangers, and the pipeline connecting the reactor core, and the primary part of the intermediate heat exchanger.

Pressurized Water Reactor (PWR) Cooling System

The reactor cooling system in a water-cooled reactor is part of a barrier that holds the cooling pressure and includes the first passive protective equipment or the first active isolation equipment. A barrier that holds the cooling pressure in the reactor with an indirect cycle such as a PWR includes the primary part of the steam generator.

The PWR reactor cooling system includes pressure vessels with top cover devices, internal reactor vessels (other than fuel assembly and core support structures), steam generators, reactor cooling pumps, pipelines that connect steam generators, reactor cooling pumps (hot leg, cold leg, and connecting lines steam generator and pump on each loop), pressurizer with relief valve and safety valve. The intended internal pressure vessel also includes the control rod directing container or fan on the pressure vessel head. The arrangement of the reactor cooling system on the PWR can be seen in **FIGURE 1** displayed below.

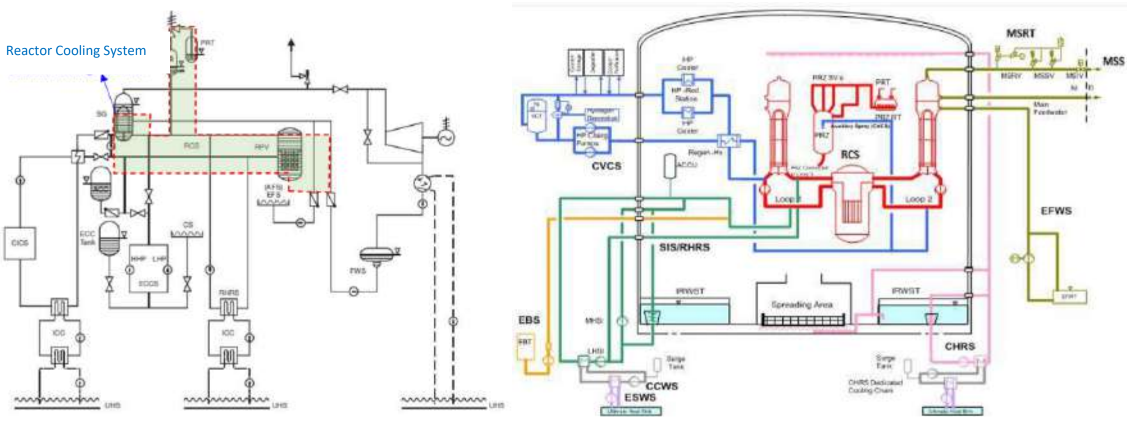


FIGURE 1. PWR cooling system and it's associated systems. [7]

Boiling Water Reactor (BWR) Cooling Systems

The barrier that holds the pressure of the reactor cooling system in the type of reactor with a direct cycle such as BWR includes the primary cooling recirculation system and the steam line and feed water line as well as the outer isolation valve. The reactor cooling system in the BWR includes pressure vessels including top devices, internal pressure vessels (other than fuel assembly and core support structures) that maintain primary cooling flow such as pumps, internal recirculation pumps, or separators. Internal pressure vessels include venturi flow, orifice, and control rod directing containers. Steam and feed water pipelines, cooling recirculation systems such as pumps, pipes, and valves are also part of the intended reactor cooling system.

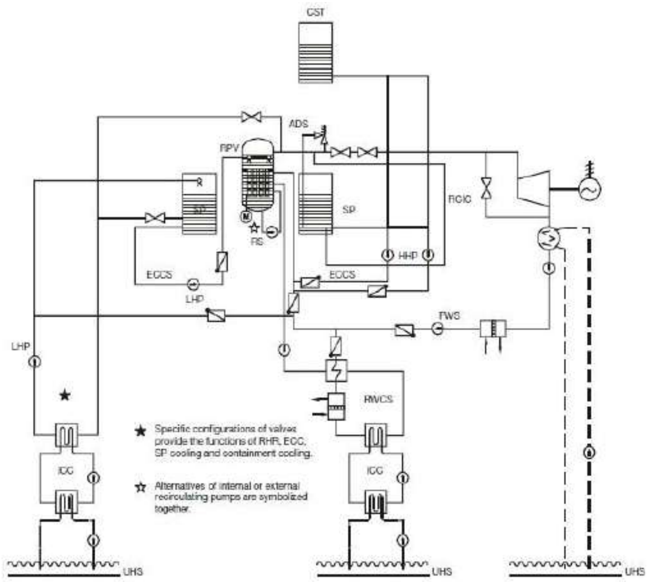


FIGURE 2. BWR cooling system and it's associated [7]

Molten Salt Reactor (MSR) Cooling System

The reactor cooling system in the MSR includes the reactor vessel, the fuel channel in the reactor core, the fuel circulation pump, the primary part of the intermediate heat exchangers, and the pipeline connecting the reactor core and the primary part of the midst heat exchanger.

The primary circulation section contains a molten salt fuel which is supplied by a centrifugal pump to the lower plenum section in the reactor vessel. The fuel solution flows from the lower plenum through the fuel channel in the reactor core to the upper plenum.

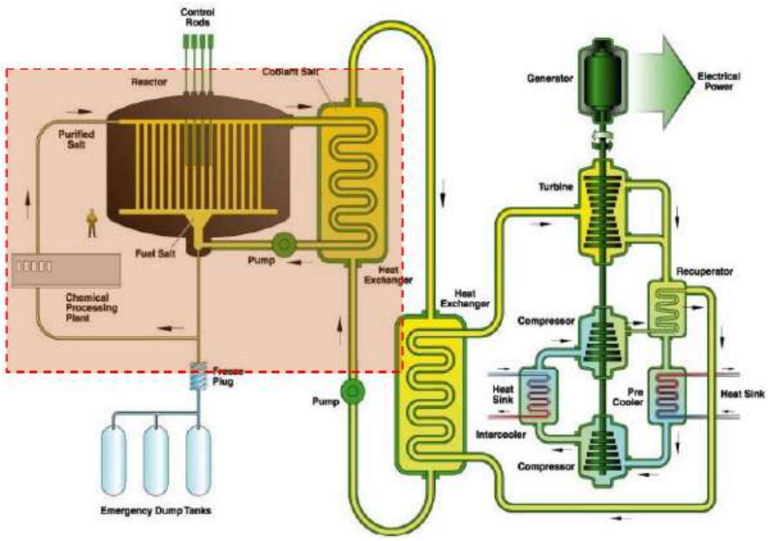


FIGURE 3. MSR cooling system and it's associated [7]

High-Temperature Reactor (HTR) Cooling System

The reactor cooling system (RCS) in the HTR-Module reactor includes reactor pressure vessels, internal reactor contents (other than pebble fuel), fuel refueling systems, steam generator, pressure vessels (other than secondary pipes), blowers, reactor ducts, and steam generators. The helium gas cooling flow is driven by a blower placed in a steam generator, pressure vessel (number 11 in **FIGURE 4**). The connecting channel between the reactor and the annulus steam generator is passed through the hot gas in the middle. The two pressure vessels (reactor and steam generator) are connected to one system (number 2 in **FIGURE 4**).

The residual heat discharges (residual heat) are designed using a surface cooler or cavity cooler placed on the primary concrete wall at the top of the reactor pressure vessel (number 13 **FIGURE 4**). The cooling capability must be able to maintain the internal integrity of the core, the extinguishing system, the reactor pressure vessel, and the primary concrete structure.

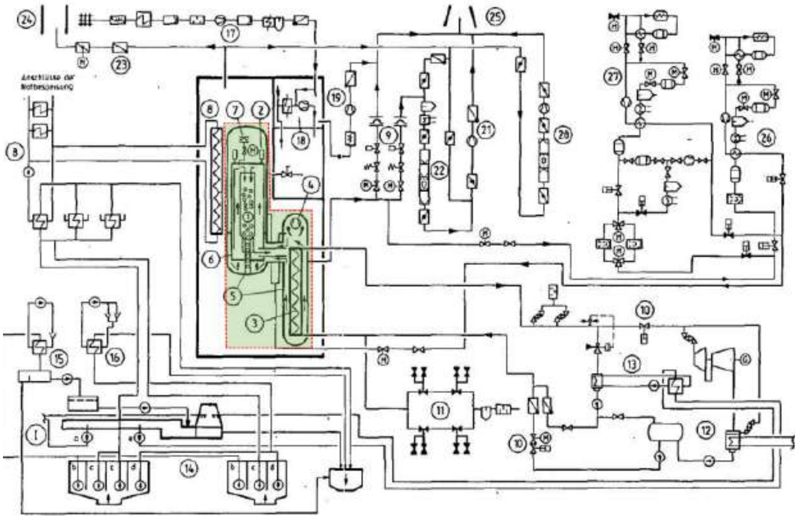


FIGURE 3. HTR-Modul cooling system and its associated [2]

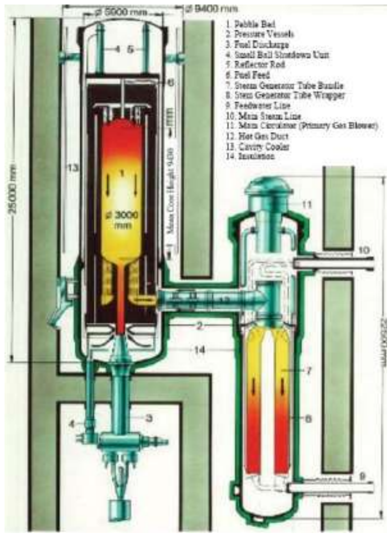


FIGURE 4. HTR-Modul cooling system [3]

RCSAS's Connected and Associated System and Ultimate Heat Sink

Connected System is a system that is directly connected to RCS (some PWR designs have a connected system that is on the secondary side of the steam generator) that is connected to the reactor cooler. The connected system serves to maintain the structural integrity of the RCS during normal operation, transients, and ultimately the design basis accident conditions. Systems that carry out the functions of safety, as follows:

a. Fluid system to control reactivity
b. Inventory, systems to control chemical for reactor cooling, including cleaning systems of reactor cooling
c. Helium cooling gas purification system (available on HTR)
d. Fission product gas discharge system (MSR)
e. The fuel processing system (MSR)
f. Emergency boron make up system (PWR)
g. Emergency core cooling system
h. Residual heat removal system
i. Drainage tank residual heat removal system (MSR)
j. Cavity cooling in the HTR
k. Main steam and feed water systems (on PWR and PHWR)
l. Makeup water systems and emergency feedwater on PWR and PHWR
m. The over-pressure relief system includes relief valves and safety valves, drain valve lines, and related equipment.
n. Heavy water collection systems for PHWR interface systems (such as sampling systems and spent fuel cooling systems) are not included in this paper.

Associated systems are the principal system for RCS and connected systems that are primarily for transferring heat to final heat dissipation. Associated systems are consisting of cooling water systems for components, intermediate cooling loops, basic service water systems, and moderator systems with their cooling systems. Associated systems are important systems for RCS and connected systems, which are generally used to transfer heat to the ultimate heat sink, such as water-cooling systems component, intermediate cooling circuits, essential service water systems, moderator systems, and cooling systems on PHWR. Heat transfer system to final heat dissipation at HTR for example heat dissipation from surface coolers using heat exchangers to cooling towers. Examples of related systems in MSR are heat exchangers for the removal of heat from draining tanks into the atmosphere using cooling towers.

The ultimate heat sink is generally sources of water, groundwater, or atmosphere which are residual heat discharging media in normal operation, anticipated operational events, or accident conditions. When water is chosen as the ultimate heat sink media, several aspects must be considered: the size of the water supply; type of cooling water supplier (for example sea, lake, a natural or man-made reservoir, or river), a water-boosting source for the ultimate heat sink, the ability of heat dissipation to provide cooling flow needed in operational conditions, accident conditions or outages in reactors.

DISCUSSION

This assessment highlighted on design requirement to be fulfilled according to the regulation. Based on the defined requirement, hence the function of the system is determined. The system function will elaborate on Structure, System, and Component (SSC) classification. Safety provisions need to be established to provide recommendations for designers and owners who will apply for a license to build and operate a nuclear reactor. The safety provisions for the design of the reactor cooling system and related systems are generally prepared based on the experience of a water-cooled nuclear power plant, due to the relatively large amount of experience in building and operating the nuclear power plant. Several technical recommendations are given for other types of reactor technology, such as the type of HTGR and MSR.

Design General Requirement

BCR Number 3 of 2011 is the only design provision available for design in general. To define safety provision for specific RCSAS, the author would like to identify how complete the BCR is. Indonesian regulation does not follow the IAEA hierarchy of the code and standard. From the gap analysis performed, any distinction on lack of requirement was identified and subject to adjust on the future regulation amendment.

General design requirements are regulated in BCR Number 3 of 2011 concerning Safety Provisions for Design of Power Reactors in Articles 40-65 which include requirements for the reliability of SSC (articles 40-48), requirements for ease of operation, maintenance, surveillance, and inspection (articles 49-52), requirements for nuclear emergency preparedness and response (articles 53-57), requirements for ease of decommissioning (articles 58), requirements for radiation protection (articles 59-63), requirements for human factors (articles 64) and requirements for minimizing aging (articles article 65). A comparison of the general design requirements in BCR 3/2011 with the IAEA SSR-2/1 document can be seen in **TABLE 1** below.

TABLE 1. Comparison of the general requirements of power reactor design frameworks in BCR 3/2011 [4] and IAEA SSR-2/1 [5]

BCR 3/2011, Article 39	IAEA SSR-2/1, Ch. 5
SSC reliability (redundancy and single failure criteria, diversity, independence, fail-safe).	R13: Categories of plant states R14: Design basis for important items for safety R15: Design limits. R16: Postulated Initiating Events (PIE) R17: Internal and external hazards R18: Engineering design rules R19: Design Basis Accident (DBA) R20: Design Extension Condition (DEC) R21: Physical separation and independence of safety systems R22: Safety classification R23: Reliability of items important to safety R24: Common cause failures R25: Single failure criterion R26: Fail-safe design R27: Support service systems R28: Operational limits and conditions for safe operation

Ease of operation, maintenance, surveillance, and inspection.	R29: Calibration, testing, maintenance, repair, replacement, inspection, and monitoring of item important to safety R30: Qualifying items for safety
Preparedness and response to a nuclear emergency (evacuation routes, markings, ventilation, auxiliary buildings).	R36: Evacuation route. R37: Communication system
Ease of decommissioning (material, access, methods & tools, waste handling).	R12: Features to facilitate the management of radioactive waste and decommissioning
Radiation protection (material, identification, fission products, work area, layout, decontamination).	R34: System containing fissile or radioactive material.
Human factors (ergonomics, manual-automatic, action time, interface, psychology).	R32: Design for optimization operator performance
Aging Minimalization	R31: Aging management
	R33: Safety systems, and safety features for design extension conditions, of units of multiple units nuclear power plant R35: Use of cogeneration. R38: Control of access to the plant. R39: Preventive access without permission. R40: Preventive of dangerous interactions with the system essential for safety R41: Interactions between the electrical power grid and the plant. R42: Safety analysis

According to **TABLE** 1 shown above, it appears to have a similar framework to the IAEA document SSR-2/1 with several deficiencies (R19, R33-42). In particular, it seems that BCR 3/2011 still needs to be more detail referring to the IAEA SSR-2/1 document (more detail and accommodates DEC, the use of cogeneration, aspects of the safety and security interface).

Design Objectives

The safety objective of nuclear reactors can be achieved with a high degree of reliability. The purpose of nuclear safety consists of general objectives and technical objectives. The technical safety objectives are explained in BCR 3 of 2011 concerning Safety Provisions for the Design of Power Reactors. According to BCR 3/2011, Article 5, Paragraph 5. The objectives of technical safety as referred to in paragraph (3) as follows:

a) prevent accidents during the operation of the power reactor and mitigate radiological impacts if accidents occur;
b) ensure with a high degree of confidence for all accidents that have been considered in the design of the power reactor will pose at the lowest risk; and
c) ensure that accidents with serious radiological impacts have very little probability.

The primary purpose of RCSAS is to ensure the availability of flow and the quality of the cooling to dissipate heat from the design basis accidents. RCSAS is to perform consequences mitigating on design basis accidents (DBA) and beyond design basis accidents (BDBA). Another purpose of RCSAS includes controlling reactivity, chemical controlling of the reactor, and heat exchange from other safety systems. This was explained in BCR 3 of 2011 concerning Safety Provisions for the Design of Power Reactors.

The fundamental safety functions of nuclear reactors must be carried out during operation, during, and after the occurrence of DBAs and accidents that exceed the specified DBA. SSC identification must be carried out following the specified postulated initiating events (PIE). The latest codes and standards must be used in the design of the SSC.

All these objectives must be fulfilled following the adequacy of design provisions. These conditions may vary depending on the type of reactor, operating conditions, and location of the plant (e.g. environmental conditions). To fulfill all of these objectives, the RCSAS design is recommended to fulfill the following functions:

- provide and maintain an inventory of reactor cooling sufficient to cool the core in all operating conditions and the design basis accident and transfer the heat generated to the ultimate heat sink.
- perform cooling flow sufficiently to meet requirements with material design constraints.
- prevent limiting the uncontrolled loss of pressure inventory in the reactor.
- maintain sufficient reactivity worth (according to operating requirements, transients, and blackouts) and prevent the insertion of uncontrolled reactivity and maintain compliance with fuel design limits

The safety objectives of RCSAS cannot be compromised when RCSAS component failures occur. The design recommendations for RCSAS focused on no initial internal and external trigger events that could initiate generator conditions that can affect the integrity of the fuel cladding or pressure limitation on RCSAS.

Safety Systems in RCSAS

The determination of a system into the safety system must be done based on the safety function in the SSC of the power reactor. Other connected systems and related systems in RCSAS are provided to mitigate the consequences of design accidents and are therefore considered a safety system. Depending on the choice of design, there is flexibility to specify various systems to carry out the required safety functions. For example, some PWR designs, additional feed water systems are used to mitigate the consequences of design accidents and are therefore grouped into safety systems. In some other designs, additional feed water systems are not used to design basic mitigation of accidents.

The determination of the safety function in the connected system and related systems may vary but each safety system in RCSAS must have the attributes to provide high confidence that it can provide safety functions:

1. The capacity as required. The system must have capacity sufficiently to perform the functions that it should and provide a high level of confidence so that fuel and RCS design limits are not exceeded. In building the required system capabilities, consideration must be given to the most dangerous conditions the system expects in operation;
2. Single failure. The system must be designed so that there is no single failure that can prevent the fulfillment of intended safety functions or other systems;
3. Electrical and emergency power sources. Appropriate emergency power (AC or DC) must be provided according to the needs of the components that need for system or operation actuation;
4. Protection over external events and internal hazards. The system must be designed and planned so that there are no external events or internal hazards in the design (such as a pipe burst or flood has the potential to prevent safety functions from being fulfilled. In particular, the capacity of the system or its components must be maintained in the most dangerous seismic conditions considered in the design;
5. Classification, codes, and safety standards and mechanical assessment. The system must be classified and designed for safety following internationally and nationally recognized codes and standards. It must be able to withstand the condition of environment and the loads that result from the anticipated operating conditions (AOO) during the life of the installation;
6. Environmental qualifications. The system must qualify for the most dangerous environmental conditions (including seismic conditions) that are expected to happen;
7. Monitoring status of the system and behavior. Monitoring of system status and readiness in normal operation must be possible;
8. Periodic testing, inspection, and maintenance on power;
9. Manual actuation. Manual actuation systems must be possible from the main control room and if appropriate from the additional control room (supplementary).

Safety functions on connected and associated systems may vary, but each safety system on RCSAS must have the following attributes to be able to provide functions, namely: sufficient capacity, single failure, electrical power supply, and emergency power supply, protection against external and internal events, safety classification, codes and standards and testing of mechanical designs, environmental qualifications, monitoring of system status and behavior, periodic testing, inspection and maintenance, manual actuation.

Safety Classification

All SSCs including software for instrumentation and control (I&C) which are important items for safety must be identified and then classified according to their functions and significance to safety.

The entire SSC must be designed, constructed, and maintained so that the quality and reliability can following its classification.

The method of classifying the safety interests of structures, systems or components is generally determined based on the deterministic method, supplemented when following the probabilistic method and technical considerations, taking into account factors such as: (1) the safety function carried out by an item; (2) the consequences of failure to carry out its functions; (3) possible items needed to carry out safety functions; (4) the time required from the postulated initiating event (PIE) or the period taken before the safety function is operated.

Safety functions and interests of at least the SSC in RCSAS that carry out safety functions must be classified:

1) Providing a retaining pressure section on the RCS when failure can cause an accident of excessive cooling loss than the ability to compensate for reactor cooling;
2) Provides protective fission products;
3) Provides heat dissipation on the core;
4) Ensure emergency core cooling (with cooling provided directly on the core);
5) Gives negative reactivity to make subcritical reactors or keep subcritical conditions.

Design Basis

Analysis of the PIE must be carried out to maintain the design basis (acceptance criteria) of the RCSAS. The structures, systems, and components of RCSAS must be designed, fabricated, constructed, installed, tested, and inspected the following codes and standards that are following the importance of the safety function implemented. The design of components on RCSAS such as pressure vessels, pipes, pumps, and valves must follow national codes and standards or international codes and standards. In designing SSCs on RCSAS that are important for safety, the calculation must consider all external hazards such as seismic, tornado, missile, flood, a storm that may occur in all operational conditions, and basic accidents.

The design basis (design conditions and requirements) for RCSAS and its components must specify:

1) instrumentation and generator control system levels assumed to function under normal operating conditions;
2) consider the functions of generating systems that operate normally;
3) the level of operator action required and the effect;
4) the level of generator protection system and reactor protection system needed to function;
5) the level of safety system needed to operate;
6) reasonable limits for malfunctions.

The most commonly used method for RCSAS design is deterministic, where the SSC will be designed following the guidance rules. The approach is generally supported by probabilistic safety assessments (PSA) whose purpose is to verify whether the plant design does not have unacceptable vulnerabilities.

Considerable consideration must be paid to the redundancy and diversity of systems and components to achieve a well-balanced design. These considerations for safety systems must be determined on the approach of deterministic such as a single failure criterion application that is supplemented by a risk approach.

Postulated Initiating Events (PIE)

A setlist of PIEs must be established in the safety analysis of RCSAS. The consequences and possibility of events must consider. For installations where preventive maintenance is carried out during operation, the consideration needs of a PIE in conjunction with vision in the safety system has to be evaluated.

Some examples of PIE that can have a significant effect on RCSAS design include:

- Break of primary and secondary pipes;
- Trip on turbines, vacuum loss in the condenser, closure of the main steam isolation valve (BWR), and steam pressure regulator failure;
- Reactor cooling failure;
- Accidental pressure relief valve opening;
- Fall of control rod (BWR), control rod withdrawal (PWR) or boron dilution accident (PWR);
- Loss of external power;
- Pipe failure in the heat exchanger on PWR (for example, a pipe burst in a steam generator);
- Internal explosion;
- Internal flooding;
- Fire;
- Earthquake;
- Explosion from outside (external);
- Floods and other natural phenomena;
- The consequences or results of human activities (including sabotage).

PIE analysis must be carried out in the safety analysis of RCSAS. In preparing the PIE list, the combination of events that are relevant to the RCSAS design must be considered under BCR 3/2011.

Seismic Consideration

The structures, systems, and components of RCSAS must be classified and applied according to a sufficient seismic category according to recommendations and guidelines. All SSC of RCSAS of safety level must be considered seismic category I if the instrument is needed to influence the following:

- Maintain the integrity of the RCS pressurized refrigerant;
- Achieve and maintain residual heat dissipation;
- Achieve and maintain reactor outage functions;
- Mitigate the consequences of seismic events.

The structure, system, and components of RCSAS must be designed based on seismic ground surface movements according to the site and seismic category according to the procedure. Barrier, supporting, and buffer devices must be provided with relevant limits on stress and shift and loss of function criteria.

The dynamic effects of flow instability and dynamic loads (e.g., water hammer) induced by earthquakes must be considered in the design according to the analysis of safety. Combinations of earthquakes and other PIE that are likely to occur independently of earthquakes must also be considered by using appropriate methodology and appropriate provisions must be made for these combinations.

Accident Events

Accident conditions that are relevant in the RCSAS design must be accidents that have the potential to cause excessive mechanical load on RCS components or cause fuel cooling is no longer be possible. Accident conditions considered for example:

- Loss of cooling accident (LOCA);
- Leakage cooling reactor to the secondary side (PWR, PHWR, HTR, MSR);
- Broken pipe in a steam generator (PWR, PHWR, HTR, MSR);
- Loss of residual heat dissipation under shutdown conditions;
- Reactivity anomaly and power distribution.

Design basis accidents must be identified and calculated for RCS to ensure the adequacy of safety system performance. The DS 481 draft document requires the identification of accidents that are included in the DEC. DEC accidents such as power station loss (station blackout), anticipated transient without scram (ATWS), total loss of feedwater system, small LOCA with failure of core cooling system, loss of residual heat discharges to final heat dissipation, loss of final heat dissipation. DEC types of accidents can be approached by accidents that exceed the design basis.

Reliability

The system needed to fulfill the safety function must have a sufficient level of confidence following the safety function implemented. An assessment of the level of system trust must consider appropriate considerations that must be given to redundancy and diversity. Redundancy as a single parameter may not be enough to provide a sufficient level of trust in seeing general failures.

Diversity could potentially compensate for this condition. In assessing the potential benefits of diversity, the following must be considered:

- The consequences of each condition operation;
- The effect of different manufacturing processes on the level of component confidence;
- The consequences on the components level of confidence for different work processes in different physical methods;
- Potential benefits and losses resulting from the increased complexity of maintenance and/or increased workload for operators in the accident.

Because redundant systems and diversity also have the potential to be vulnerable to events (for example fire, flood) resulting from common failures. Appropriate physical protection or physical separation or a combination of the two must be used as far as practically possible.

Material Selection

The material used to maintain the pressure barrier in RCSAS must be compatible with existing cooling, with materials used for jointing (for example welding materials), as well as adjoining components or materials such as sliding surfaces, piles of boxes, and radiolytic products.

The material specified in RCSAS must meet with the provisions applicable to the code used, including but not limited to the following properties and characteristics:

- Resistance of heat;
- Tensile, cracked, and fatigue properties;
- Properties associated with corrosion and erosion;
- Resistance to cracks caused by corrosion stress;
- The irradiation effects resistance;
- Resilience to fragility;
- The characteristics of ductility (including flattening growth speed);
- Characteristics of fracture toughness (brittle failure);
- Fabrication easiness (including welding);
- Resistance of reactions on metal-water.

Materials must be defined to fit the service conditions expected in the overall operating conditions and the basic accident conditions of the design.

Over Pressure Protection Provision

All pressure-bearing components in RCSAS must be protected from overpressure conditions following applicable codes and standards. All pressure-bearing components in the RCSAS must be designed following the sufficient safety margins (margin of safety) to ensure the pressure limits are not exceeded and the limitation of fuel design does not exceed operational conditions or in DBA.

The design of the RCS must include features sufficient to protect against overpressure, that is, the feature must provide the ability to deal with vapors and solutions in RCS. Relief valves and/or safety valves must be included in the design. The multi-layered defense concept must be applied to protect overpressure. The principle of diversity must be applied to the design of overpressure protection on RCS to reduce the possibility of general failure.

The design of the overpressure protection equipment must reflect the significance of safety and must be consistent with the performance expectation at the PIE limit in general.

Overpressure protection at the reactor cooling pressure limiter can be achieved by methods which follow the provisions:

- System pressure monitoring;
- Method of regulating system pressure in operational limits (example: use inventory control systems);
- Equipment to relieve pressure such as valves for safety and relief;
- Reactor protection system (RPS).

List of methods on reducing and / or managing pressure on RCS as follows:

- Spraying methods inside a pressurizer (in PWR);
- Open the pressurizer release valve for PWR and the discharge valve on the pressurizer for PHWR
- Open the safety valve;
- Open a bypass valve on the turbine.
- Open the release valve on the main steam path;
- Initiating trip reactors by the reactor protection system;
- Prevention of excessive cooling injection (for example when RCS operations are conducted with an isolated pressurizer while initial warm-up transient or transient at PHWR);
- During startup or shutdown of the reactor, the reactor cooling discharges through RCSAS, or at PWR using the let-down function on the system to control inventory and chemical.

In the design and location of safety valves and/or relief valves in RCS, pressurizers (PWR), and related vessels (if possible), consideration must be taken in the single failure criteria so that the limits of RCS always be achieved under limits of design in operational and DBA.

The capacity of discharge of the safety valve and/or relief valve in the RCS must be sufficient to limit the pressure rise and maintain the pressure within the design limits that have been determined during operating transient conditions and in accident conditions by considering the design on the RCS, following the codes and pressure standards that can be applied. The number of valves must be sufficient to provide the required level of redundancy.

MSR Gas Release Management System

Volatile fission products such as xenon and krypton must be continuously taken from the reactor. The gas must be managed properly before being discharged after sufficient decay. The off-gas system can be seen in **FIGURE 5** (in the box).

The intake of volatile gas from the primary cooling system can be done using the helium gas bubble method (as shown in **FIGURE 5**). The flow diagram of the gas separation process starts from the gas discharge from the primary system to the storage in a cylinder described in **FIGURE 6**.

The fission product gas will initially be released in the drain tank. Furthermore, fission product gases (xenon and krypton) will pass through the activated charcoal filter. Most of it is separated from other gases by cryogenic separation. The separated gas is then stored in a pressurized cylinder until sufficient decay.

Layout Consideration

The layout design of the RCSAS must consider radiological protection from personnel, protection against the pipe failures consequences, protection against internal missiles, provisions for ventilation, and drainage of the reactor cooling, and provisions to accommodate activity of testing and inspection.

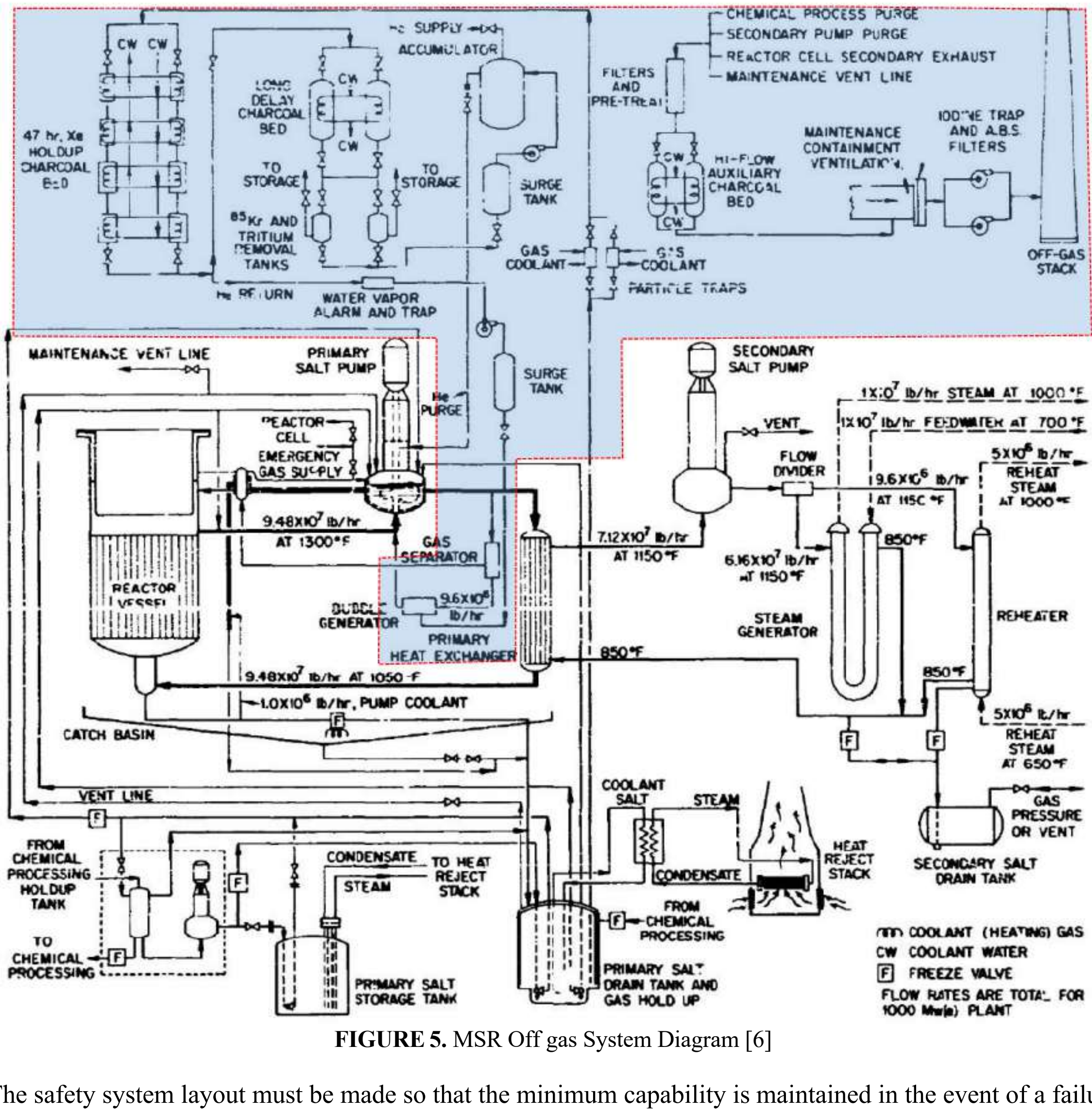


FIGURE 5. MSR Off gas System Diagram [6]

The safety system layout must be made so that the minimum capability is maintained in the event of a failure in a protection scheme or in events that require safety from internal and/or external hazards (e.g. earthquake, fire, and flood). Spatial and drainage requirements must be considered and the provisions for this must be following the level of maximum on external flooding at the site installations

The RCS layout must be made so that in power supply failure or loss event for the pump, in operational conditions or specific conditions of design basis accident, residual heat discharges are maintained by using natural circulation from the reactor cooling.

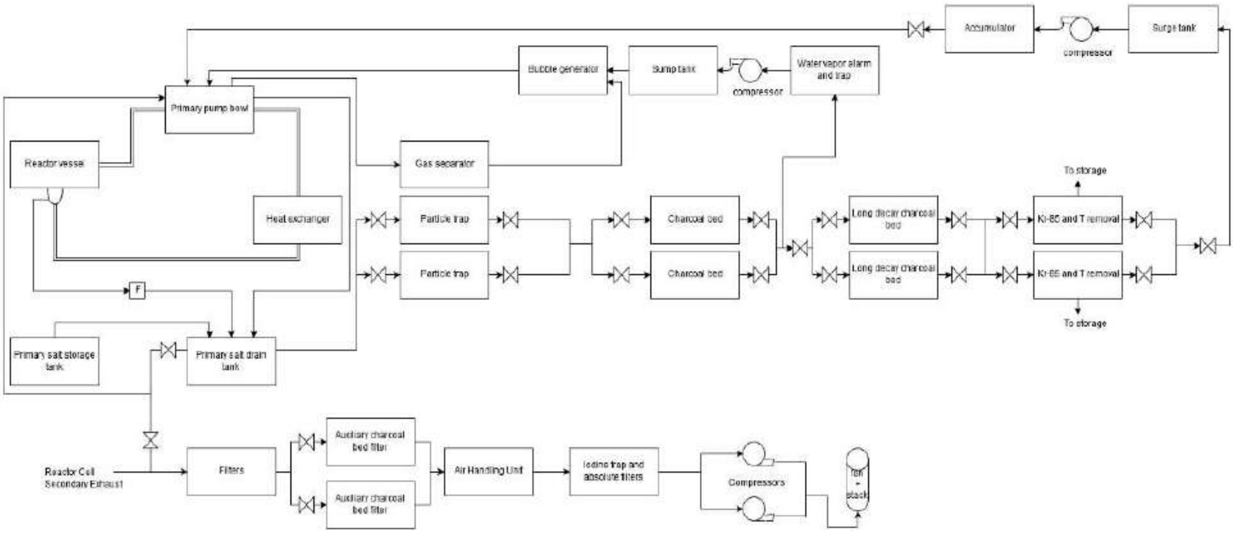


FIGURE 6. MSR Fuel Management Diagram [6]

Protection against Radiation Exposure & Pipe Damage Consequences [8]

The layout design of the RCSAS must allow for the inspection, maintenance, repair, and handling of the SSC by considering personnel radiation protection. The things that need to be considered radiation protection objectives, as described below:

- Circulating contaminated water system and components must be equipped with adequate shielding of radiation;
- The part of the connected system that is located between the RCS and the first isolation valve, including the valve, which is normally closed during normal operation must be designed with the same safety standards of RCS;
- System of fluid that gets into the enclosure to the outside of the confinement must be robust and equipped with sufficient equipment to isolate the flow that can maintain the safety and performance functions of the confinement. Parts that penetrate the enclosure and include a flow barrier must be considered an extension of the confinement and are designed according to the level of quality and performance following applicable codes and standards. The system, if not equipped the ability to detect as well as isolate quickly from leaks, must be considered an extension of the confinement barrier and must follow the design;
- Systems of fluid that interacts with circulating contaminated water systems and components must be designed to prevent and minimize leakage. The leakage from radioactive products can be prevented or can be detected quickly;
- A long pipes lines carrying radioactive material must be minimized in areas where personnel might be exposed to radiation;
- Other gaps and local configurations where deposits and debris of radioactive can accumulate must be reduced and minimized in detail designs on the plant.

Design actions for radiation protection can be more clearly spelled out in IAEA standard documents on aspects of nuclear radiation protection [9].

Consideration must be given to the system of the piping layout of as well as the design of the pipeline supports to protect the SSC from the pipe damage consequences. The specification of RCSAS design must identify pipes with high energy when sudden bursts are postulated to occur and systems that must be protected from dynamic effects such as pipe bursts.

Interface Consideration

Appropriate interface equipment must be provided for connections between systems and components that have different safety classifications. Some examples of interface equipment such as pipes in the heat exchanger (on PWR, HTR, and MSR) that separate two different fluids, valves that are operated remotely (which maintain systems with higher safety functions), manual valves, equipment to block flow (e.g. orifice which limiting the cooling flow rate on RCS). Interface equipment must prevent loss of function on system and component with a higher safety classification and prevent the release of radioactive material. Interface equipment must have the same safety classification as systems and components with higher safety classifications connected.

In the RCSAS structural design, consideration must be considered on the effect of the overall safety of the plant installation. The designer of the plant must make sure that the structure's temperature and the interfaces in the RCSAS will remain inside the limit of acceptance and conditions are made by in-service inspection. Components and structures that are directly connected to the containment must be designed. Hence the failure will not result in loss of the ability to leak resistance from the confinement.

Isolation Consideration

Adequate isolation must be provided at the interface between the RCS and the connected system which operates at lower pressure to prevent excessive pressure on the system and the possible loss of a cooling accident. Consideration must be given to the characteristics and importance of isolation and the level of confidence of the target. Insulation equipment must be closed or closed automatically according to request. The response time and closing speed must be adjusted to the acceptance criteria defined from the initial trigger event.

Instrumentation and Control System

An instrumentation and control system with a safety grade must be available for activating the appropriate safety system and provide information sufficient for the reactor operator to aware of the determination of the RCSAS state. The instrumentation and control system (I&C) must also be able to monitor plant conditions during normal operation and for anticipated events.

Inspection, Test, and Maintenance Consideration During Operation

The structure, system, and components of RCSAS design facilitate the performance during inspection and testing without unnecessary exposure to personnel on-site. Appropriate in-service inspection programs must be developed for the whole lifecycle of the plant and also for the commissioning period.

Structures, systems, and important components for safety must be inspected during the service period by looking at the ability to perform the intended safety function and physical integrity, including any changes in property and characteristics of the material used. Specified inspections and testing methods should not require the ability to carry out inspections and tests that exceed techniques and methods that have been developed and accepted.

Equipment that operated automatically or remotely can be used for inspections during operation to maintain the inspection personnel exposure as low as possible and within the limits set by the regulatory body.

Advanced Reactor Design

The identification and evaluation of the differences in key design features in the design of advanced reactors against LWR and Heavy water reactor (HWR) generation reactors must be considered in the assessment related to the applicability of the guidelines for the design of the advanced reactor.

Reactor designs with significant differences from the current reactors must be sufficiently tested ensuring that thermal-hydraulic behavior is determined and predicted. Data analysis and code tests must be initiated. The code must be provided to predict reactor behavior in transient and accident analysis.

SUMMARY

General design requirements for RCSAS general provisions can be applied to the components of RCSAS. The majority of requirements can be applied to various types of reactors. Some requirements only apply to certain types of reactors according to the design uniqueness, for example, a fission product gas system (only on MSR). Provisions regarding the accumulation of combustible gases only apply to water reactors (PWR, BWR, PHWR) and do not exist in HTR and MSR type reactors.

All those descriptions on the general design requirement on RCSAS are acting as recommendations to develop safety provisions on RCSAS regulation by BAPETEN, hence the regulation is useful to provide recommendations for designers and owners who will apply for a license to build and operate a nuclear reactor in the future.

ACKNOWLEDGMENTS

This study was a joint assessment in collaboration with P2STPIBN-BAPETEN and the Department of Nuclear Engineering and Physics Engineering UGM in 2019. The paper discussed only a number of topics from the study due to paper limitation. The authors would like to thank you to all relevant parties for the fruitful cooperation as well as the hard work and effective work flow until the end of activities.

REFERENCES

[1] IAEA Safety Guide No. NS-G-1.9, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants, 2004.
[2] Breitbach, G., David, P. H. et. Al., Zum Stoerfallverhalten des HTR-Modul Eine Trendanalyse, Spezielle Berichte der Kernforschungsanlage Juelich, 1984.
[3] AREVA, Technical Data Record, Pebble Bed Reactor Plant Design Description, 2010.
[4] BAPETEN Decree No. 3, Safety of Nuclear Power Reactor Design, 2011
[5] IAEA SSR-2/1, Safety of Nuclear Power Plants: Design, Specific Safety Requirements, 2012.
[6] Robertson, R. C., Conceptual Design Study of a Single Fluid Molten Salt Breeder Reactor, ORNL, 1971.
[7] IAEA Draft DS 481, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants, 2017.
[8] Assessment Report on Design Safety Provision on Reactor Cooling System and Associates System, P2STPIBN-BAPETEN and UGM, Jakarta, 2019.
[9] IAEA Safety Standards NSG 1.13, Radiation Protection Aspects of Design for Nuclear Power Plants]
[10] Government Regulation No. 54 on Nuclear Installation Safety and Security, 2012

Lesson Learned from Regulatory Review and Assessment for Periodic Safety Review of 30 MW MTR Type Research Reactor in Indonesia

Rahmat Edhi Harianto[1)], Budi Rohman[2)], Wiryono[3)]

Directorate of Licensing for Nuclear Installation and Materials - Nuclear Energy Regulatory Agency
Jalan Gajah Mada No. 8 Jakarta, Indonesia
[2,3)]Nuclear Energy Regulatory Agency, Indonesia

Correspondence email: r.eharianto@bapeten.go.id; rahmatedhih@gmail.com

Abstract. Lesson Learned from Regulatory Review and Assessment of Periodic Safety Review for 30 MW MTR Type Research Reactor in Indonesia. The first periodic safety review (PSR) for the Indonesian 30 MW MTR-type research reactor (RSG-GAS) completed meeting the mandatory requirements. RSG-GAS obtained an operating license from 1995 to 2020. During the operation of the reactor, BAPETEN and BATAN performed several inspections and examinations to ensure the safe condition of the installation. But those inspections performed partially, not related to each other, and have not reflected the entire comprehensive examination like periodic safety review (PSR). BATAN has submitted a collection of documents to BAPETEN includes the PSR document in the year 2017. The establishment and submission of PSR by BATAN as the operator and review and assessment performed by the BAPETEN as the regulatory body are the first events for both institutions. There has never been a real example or model procedure in the implementation of PSR before from both sides. The research of methodology uses a descriptive and comparative approach. The paper is a literature study of the formal process of PSR that compares regulatory review and assessment of PSR in several countries that regulate a research reactor, IAEA standards, and Indonesia regulation. The study concludes that PSR implementation in several regulating research reactor countries is varied. PSR regulations in Indonesia are sufficient. However, it needs updating specifically regarding the aging management program in GR 2 of 2014 and PSR formal procedure in the BCR or nuclear reactor licensing procedure. The results of PSR give benefits and lessons for the licensee and the regulatory body. BAPETEN gains benefits as follows: an input to update the regulation and procedures and develop databases for the licensing decision support system. Meanwhile, BATAN gains lessons as follows: to know actual plant safety, also to plan and to set work priorities for safety improvements.

Keywords: periodic safety review, regulatory review and assessment, research reactor

INTRODUCTION

RSG-GAS was built in 1983, that the first criticality attained in 1987, after due approval of the regulatory body on the first core operation. RSG-GAS operates on 30 MW was achieved on the 6th core in 1992. RSG-GAS obtained an operating license from 1995 to 2020. RSG-GAS is a research reactor in which the utilization for the radioisotope production, material irradiation, material testing reactor, neutron activation analysis, neutron beam utilization, and education and training[1].

In the operation reactor, BAPETEN conducts routine inspections three times a year to monitor compliance with the license[2]. The licensee also performed an in-service inspection to identify the degradation of the structure, system, and component in aging management framework implementation, and stress tests after the Fukushima Daiichi accident. The purpose of the stress test is to examine the capability of installation in response to natural or external hazard in the site such as seismic and volcanic includes deterministic safety analysis with several scenarios such as loss of flow accident, loss of off-site power, and reactivity insertion accident due to inadvertent control rod withdrawal[3].

However, those inspections performed partially, not related to each other, and did not reflect a comprehensive examination such as periodic safety review (PSR). RSG-GAS has never been performing of PSR due to it is not mandatory according to the regulation.

Recently, PSR implementation is obligated. The PSR document included in the technical document requirement for operating license renewal based on Government Regulation (GR) No. 2 of 2014 on Licensing of the nuclear reactor and nuclear materials utilization [4]. In December 2017, BATAN submitted a collection of documents to BAPETEN includes the PSR document.

The establishment and submission of PSR documents by BATAN as the operator and review and assessment performed by the BAPETEN as the regulatory body are the first events for both institutions. There has never been a real example or model procedure in the implementation of PSR before from both sides.

This paper is a literature study of the formal process of PSR consisting of regulatory review and assessment of PSR in several countries that regulate a research reactor, IAEA standards, and Indonesia regulation. The availability and applicability of Indonesia PSR regulation such as GR and several Bapeten Chairman Regulations (BCR) are studied, also the benefit and feedback of the PSR implementation.

Overview of PSR Regulation

International best practice

Each country applied PSR provisions differently. Generally, the PSR applied in power reactors but in some countries also expect the PSR implementation in the research reactor. The following are the PSR practices in several countries in **TABLE 1.**

TABLE 1. The Implementation of Research Reactor PSR in Several Countries.

Country	USA	Australia	Netherlands	France	Germany	Slovenia	South Korea	India
Power	0.01 watt - 20 MW	> 10 MW	50 MW	N/A	20 MW	250 kW, pulsed 1 GWT	30 MW	100MWt
type of reactor	MTR & TRIGA	OPAL & HIFAR	HFR	N/A	N/A	TRIGA	MPR	N/A
Regulation of PSR and frequency	Not require	Every ten years (*)	(*)	mandatory	(*)	prerequisite of renewal operating license)	(*)	(*)
Reference of PSR	-	IAEA SSG-25 (**)	(**)	N/A	(**)	NSG 2.10 (previous of **)	(**)	NSG 2.10 (previous of (**)
Scope of PSR	-	With a graded approach (***)	(***)	(***)	Full PSR (+ PSA)	(***)	(***) & Full PSR (+ PSA)	Full PSR (+ PSA)

Recently, the Nuclear Regulatory Commission of the United States (USNRC) regulates 31 operating research and test reactors. Most research and test reactors in the USA are at universities or colleges with range power in size from 0.01 watt to 20 megawatts-thermal (MW(t)). The technical document for license renewal is the full scope of FSAR to the facility with licensed power levels of 2 MW(t) or more. Meanwhile, the facilities with licensed power of less than 2 MW(t) perform a review that focuses on the most safety-significant aspects of the renewal application and considers past NRC reviews. Most significant to safety are radiation protection, waste management programs, financial requirements, reactor design and operation, accident analysis, and technical specifications [5].

In 2019 USNRC published new requirements in 10 CFR 50.71(e) for non-power and utilization facilities (NPUF) licensees to submit to the USNRC an updated FSAR and subsequent FSAR updates at intervals not to exceed five years. This change is motivated by the observation of license renewal applications for more than 20 licensees during the period 2006–2017. Some licensees did not adequately update their FSARs or did not properly maintain the supporting references. This event often led to delays in the license renewal reviews and significant resource expenditures to both licensees and the USNRC. According to NPUFs requirement, the information consists an evaluation of 1) a change made to the facility or a facility major modification, 2) a change in an SSC as part of major preventive or corrective maintenance (e.g., replacing an analog meter with a digital readout, replacing a safety-related pump with one that has increased flow); or 3) change in the facility, procedures, or experiments not previously described in the FSAR, should be considered in the FSAR.

The others information is evaluations regarding potential or actual aging of SSCs and any aging management actions taken, changes in the facility site environments, for instance: new industrial, transportation, military, or residential facilities near the facility site or changes in the population potentially exposed to the facility releases), and changes significantly of design basis in the facility site environs related to natural phenomena, including geography, meteorology, geology, hydrology, and seismology[6]. However, there is no requirement for PSR in the USA research reactor regulation.

The PSR requirement practice in the USA is different from Australia, France, Netherlands, and Slovenia. In the four later countries, the PSR has become a mandatory requirement to be implemented every ten years. Moreover, in Slovenia, PSR has become an operating license renewal prerequisite. However, the adaptations and graded approaches of PSR apply due to specific features of research reactors in those countries [7-10].

Further, Germany, South Korea, and India employ PSR provisions with a more rigorous requirement. A research reactor in those countries has to perform a regular PSR every ten years with full-scope safety factors includes probabilistic safety analysis [11-13].

IAEA Standard

The formal process of implementing PSR on power reactors is accomplished in four stages [14], as shown in **FIGURE 1**. The first stage is the preparation of the PSR project to establish a project team involving both the operating organization and the regulatory body. Then, the operator and regulatory body discussed the scope, level of detail, the timing of the review, and the codes and standards that will be employ. The summary of the discussion resulted in an agreement in the form of a PSR 'basis document'. The second stage is conducting the PSR. In this stage, the operating organization performs an agreed 'basis document' for the PSR. The review includes identifying positive or negative findings and a global assessment report (GAR) of the facility. These negative findings may lead to corrective actions or safety improvement proposals.

The third stage is performed by a regulatory body that reviews and assesses the PSR report prepared by the operating organization. The regulatory body then proposes corrective actions or safety improvements. Later, the regulatory body identifies any safety issues in each safety factor to makes categorization and prioritization of the safety improvements.

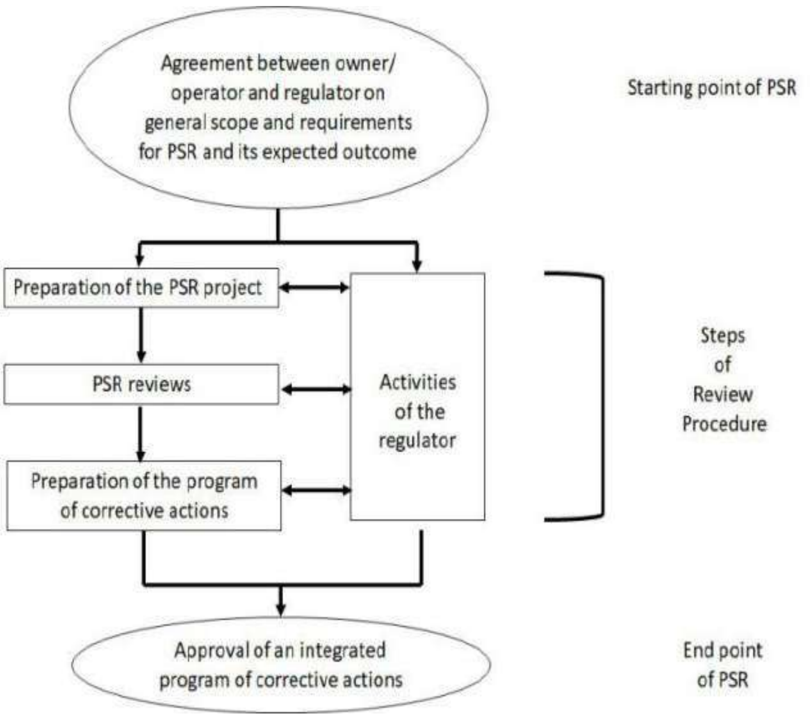


FIGURE 1. The overall process for a PSR of a nuclear power plant.

The last stage is the finalization of the integrated implementation plan. The integrated implementation plan has to contain corrective actions also reasonable and practicable safety improvements as needed, along with a schedule approved by the regulatory body.

METHODOLOGY

The research methodology carried out using a descriptive and comparative approach. This paper describes the formal process of PSR comprise regulatory review and assessment of safety factors in several countries regulate research reactor, IAEA standard, and Indonesia regulation. For Indonesia regulation, GR No. 54 of 2012 on The safety and security of nuclear installation [15], and several Bapeten Chairman Regulation (BCR) are used to acquire information relating to the review and assessment of PSR documents. This paper will answer several research questions:

1. How does the implementation of PSR in other countries?
2. How the availability of regulation on PSR regulatory review and assessment of the research reactor in Indonesia?
3. What is the benefit and lessons gained from PSR review?

There are seven safety factors related to PSR document comprises of 1) plant design, 2) actual condition of important the structure, system, and component (SSCs) to safety, 3) qualification of important SSCs to safety, 4) aging, 5) safety performance and use of experience from other plants and research findings, 6) organization, management system, procedures, and emergency planning, and 7) environment radiological impact. Due to the broad scope of PSR, only safety factor numbers one to five that will be discussed in the paper because only those factors have quantitative data to be input and analyze using the computer database. However, the other two safety factors contain the program and standard operating procedure of the research reactor installation.

RESULTS AND DISCUSSION

Indonesia PSR Implementation

Recently PSR provision is specified in the GR No. 2 of 2014 and GR No. 54 of 2012 that is set more detail in BCR No. 2 of 2015 on Safety Verification and Assessment of Non-Power Reactor[16]. The licensee shall periodically conduct verification and assessment of safety at several stages of the nuclear installation license includes construction, commissioning, and operation stages. The scope covers are assessments of plant design, the actual condition of important a structure, system, and component (SSCs) to safety, qualification of important SSCs to safety, aging, the safety performance, use of experience from other plants and research findings, organization, and the management system, procedures, emergency planning, and radiological impact to the environment.

As a preparation of PSR, BAPETEN and BATAN held a focus group discussion in September 2016. Both parties agreed that the PSR basis document composed following the format and contents from BCR No. 2 of 2015 with consideration of the latest of the condition of reactor assuming and analyzing for 30 MW power. The data and analysis are collected and evaluated within ten years (2005-2015) in-line stated in BCR 2 of 2015. The data of PSR taken after the RSG-GAS license operation issued in 2005 [17]. The scope, level of detail, timing of the review, the regulation, codes, and standards to be used in the PSR agreed by both parties.

Based on the focus group discussion resulted in September 2016, PRSG-BATAN composed the PSR document which the scope and content referred to BCR No. 2 of 2015 that consists of ten sections as follow: an introduction, plant design, actual condition of the structure, system, and components, equipment qualification, aging management, safety performance and operating experience feedback, safety management, and emergency preparedness, and radiological impact on the environment, conclusion, recommendation and follow-up action.

Document of PSR submitted to BAPETEN by the end of 2017. The review of each safety factor includes identifying positive/strengths or negative/deviations findings. Negative findings may lead to proposals for corrective actions or safety improvements to be addressed in chapter ten of the PSR document.

The following are five of seven safety factors of the PSR document. The consideration of essential applicable input and output reviewed.

Plant Design

Review: RSG-GAS was designed and constructed with current standards and guides for a nuclear reactor at that time. The plant design requirements fulfillment stipulated in GR No. 54 of 2012 consist of general and specific requirements of the design. General design requirements consist of design for a) The reliability of the structure, system, and component; b) The simplicity for operation, inspection, maintenance, and testing; c) Nuclear emergency preparedness and response; d) The ease for decommissioning; e) Radiation protection; f) Human factor, and; g) minimize aging.

Nuclear reactor specific design requirements consist of design for a) reactor core; b) heat removal system; c) shutdown system; d) reactor protection system; e) technical safety feature; f) containment system; g) instrumentation and control system; h) handling and storage of nuclear fuel system; i) radioactive waste system; j) auxiliary system.

All the content from general and specific design compared with the article clause in BCR No. 1 of 2011 on the provision of design safety for a research reactor[18]. Each requirement of this BCR has been examined and observed to fulfill the implementation in GA Siwabessy reactor. There are no changes in design installation since RSG GAS obtained the last operating license. The modification program of reactor fuel type from Uranium-Oxide to Uranium-Silicide has been updated and stated in the previous SAR.

From 2014 through 2016, BATAN performed site characteristic evaluation in the Serpong area by concerning RDE development. The results of Peak Ground Acceleration (PGA) is 0.57g for 10.000 years recurrence [19]. This PGA will affect the adjustment of the reactor building and structure. As a result of the assessment, PRSG-BATAN will organize a seismic assessment team to perform analysis reliability of the building and the structure, also reactor core design. Additionally, based on their result/assessment, PRSG-BATAN will develop a work program and including an engineered solution if it is needed. Another specific design required special highlighted is the design of beam tube utilization and modification.

Conclusion: From all assessment results above, it concludes that all general and specific design safety requirements can fulfill, and the adequacy of the design still meets the applicable standard. The building and structure design include the reactor core, as well as the utilization and modification design required further analysis include an engineered solution.

The Actual Condition of Important SSCs to Safety

Review: The purpose of these safety factor study is to decide the actual condition of important SSCs to safety and to estimate whether they are capable and sufficient to meet design requirements, at least until the next PSR. The study also should check that the condition of important SSCs to safety documented accurately includes the continuous maintenance, surveillance, and in-service inspection program.

The review of the present condition of important SSCs to the safety of the research reactor also represents an examination of the following aspects for each SSC that is existing or anticipated aging processes, and operational limits and conditions.

The scope of SSCs is limited to important SSCs to safety, namely the systems of buildings and structures, reactor core, matrix and fuel assembly, reactivity control including control rod, primary loop, emergency cooling, ventilation, reactor protection, safety-related of control and instrumentation, electrical power supply, handling and storage of nuclear fuel, radiation protection, and fire protection.

To determine the actual physical condition of SSCs, the reviewer examines operation reports consist of records of maintenance covers calibration, surveillance, and functional test, and inspection report. The reviewer founds a discrepancy between the measurement and calculation of control rods reactivity. Due to this discrepancy, PRSG-BATAN is committed to adjusting core management calculations using a more reliable computer code. The assessment result also estimates that the reactor protection system considered replacement within the next 5-10 years.

Conclusion: The review concludes that the plant systems, structures, and components are functioning as expected. The core management aspect requires further investigation and engineered solutions.

Qualification of Important SSCs to Safety

Review: The purposes of the qualification important SSCs to safety review is to decide whether equipment to safety has been accurately qualified (including for environmental conditions) and whether qualification maintained through a sufficient program of maintenance, inspection, and testing that assures the performance of safety functions until at least the next PSR.

The review should also organize the requirements for performing safety functions while subject to the environmental circumstances that could remain in normal conditions and predicted accident conditions. These should include seismic, vibration, temperature, pressure, irradiation, corrosive atmosphere, and humidity conditions.

To determine the equipment qualification of SSCs, the reviewer examines operation reports consist of maintenance records, calibration, surveillance, and functional test, inspection report, and the measured value on the equipment (system and component). Then the reviewer will compare this measured value with the standard design value of a system and component-specific.

Conclusion: The review concludes that these systems and components to safety performed as expected.

Aging Management

Review: The most significant safety factor for operation installation is the management of aging. The goal of the aging management review to determine whether the aspects of aging influence important SSCs to safety maintained adequately. Additionally, another purpose of this review is to decide whether effective management of the aging program is suitable so that all required safety functions will perform for the plant design lifetime or long- term operation.

In this aspect, the reviewer examines 1) the list from all important SSCs and critical of SSCs, specifically with SSCs fulfill three criteria: important SSCs to safety, not redundant, and not easy to repair/replace[20]. For instance, the reactor tank, structure building of the reactor, heat exchange, and the primary coolant pipe of the system; 2) the aging mechanism for SSCs, 3) its impact on safety function, and 4) the results of the in-service inspection. Evaluation of the aging degradation performed by comparing the condition of critical SSCs with report data from in-service inspection, surveillance, functional test, operation report, and performance indicators.

However, there is one component part of the critical SSC that cannot be observed due to the unavailability tools needed at PRSG-BATAN. For that reason, PRSG-BATAN commits to complete all assessment of critical SSCs and address it in an integrated implementation plan for safety improvements. Moreover, BAPETEN applies another data to justify that although these critical SSCs are unable to be assessed within this period, however, there is no abnormality data operation report during 2005-2015 related to this component.

The assessment result also showed that due to long-term exposure of radiation, temperature, humidity, and other environments, an aging process could happen in the component of the electric cable. Therefore, the management of the G.A. Siwabessy reactor should give attention to update the aging management program and maintenance program in the next reactor operation activity because of surveillance and maintenance of the electrical cable not included in the maintenance repair manual of RSG-GAS.

Conclusion: The review concludes that an update of the aging management program and maintenance program needs to apply in the next operation. The assessment of one critical of SSCs will be address in an integrated implementation or safety improvement plan.

The Operating Experience Feedback and Safety Performance

Review: The safety performance review aims to recognize any need for safety improvements based on the reactor safety performance indicators and records of the experience operation, the plant-related event evaluations, and root causes. The reviewer evaluates safety performance using the operation installation report data during the period 2005-2015 contains the safety-related events, and records of the safety systems unavailability, the doses of radiation, and the radioactive waste production and radioactive effluent discharge. The unavailability system is the availability of the safety system cannot function when needed.

Data operation of the reactor from core 52 to 89 showed that generally, the RSG-GAS protection system functioned properly. However, there was one event during low power reactor operation where the trip from normally power supply did not operate. There was also know where the control rod automatically falls event caused the reactor scrammed frequently. However, the evaluations of these events have been resolved. As a result, since the core of 93 in 2017, the RSG-GAS system starts in a normal condition.

Regarding radiation protection, PRSG-BATAN monitored gamma and neutron radiation exposure in a normal operating cycle. The radiation doses from core 52 to 89 in the working area of the RSG-GAS reactor generally showed a value below ten μSv/hour. The radiation dose exposure assessment of the reactor at the 8th and the zero levels operates in 15 MW showed a dose rate that is slightly above the dose limit value (NBD)/hour. This event is due to the presence of other radiation sources besides the reactor core, including a delay tank and a temporary storage area for radioactive materials at the 8th level of the reactor. However, the radiation protection implementation and the safety program document worked well, proved by the results of monitoring the receipt of the maximum personal dose for radiation workers is relatively far below the NBD determined in the regulation.

The second aspect of this chapter is the operating experience feedback of safety factor review. This factor purposes to assess if there is adequately relevant experience feedback from other research reactors and research findings, and to identify a good practice, lessons learned, and take advantage of improved knowledge derived from those researches. PRSG-BATAN highlighted several important events as feedback for future reactor operation such as 1) fission product of molybdenum (FPM) irradiation failure; and 2) some operating experiences, such as a) black spot on reactor fuel element; b) primary water conductivity raised; c) FPM leak event; d) fuel placement error events in reactor core; e) primary pump clutch broke; f) beam tube event, and g) unplanned shutdown event due to unbalanced loading of JKT03 detector.

Conclusion: The review concluded that the safety system was functioning as expected, the event of the unavailability system was followed up and resolved, and the radiation dose below the dose limit value. PRSG-BATAN management is also very well aware of operating experience reviews.

The next step activity after review completed in each aspect above has performed an analysis of the interfaces between the various safety factors with considering all the findings from each safety factor review and what safety improvements. The advance analysis or follow up action addressed in GAR also in an integrated implementation plan of proposed safety improvements linking safety significance and prioritization. The global assessment of all safety factors of the G.A. Siwabessy reactor had formulated into fourteen actions in the integrated implementation plan along with the year to resolve. The integrated implementation plan and summary report submitted and approved by BAPETEN at the end of 2019.

Lesson Learned from the RSG-GAS PSR Document Review and Assessment

The RSG-GAS PSR documents review and assessment carried out using GR No. 54 of 2012 and several BCRs. The format, content, and scope of the PSR reviewed using BCR No. 2 of 2015. In this BCR, the formal procedure to perform a PSR is not stated, specifically in the preparation of the PSR project. This step important because it will produce guidance for both parties in the form of a PSR 'basis document' concerning the scope, level of detail, the timing of the review, and the codes and standards that will be employ. However, BAPETEN already performed this stage and declared in the preparation of the PSR process above by adapting the IAEA standard procedure and similar methods in other implementing PSR research reactor countries [7,12-14].

The PSR basis document is a necessary tool that directs the process of the PSR. It assures that the licensee and the regulatory body have equal expectations for the PSR's scope, methodology, and results. The PSR basis document covers some imperatives components such as 1) description of a current licensing basis, including exclusions and tolerable deviations, 2) description of the proposed operating plan of the facility, 3) the description of the PSR scope, 4) information on the methodology for the performance of the PSR, including the period for which the PSR is valid, 5) information about applicable regulations, codes, and standards, 6) the methodology for the identification, dispositioning, and tracking of gaps, 7. the method for the GAR, and PSR administration.

Considering the importance of this process before performing PSR review and assessment, therefore the process of producing guidance for both parties in the form of a PSR 'basis document' can be taken as a lesson learned and a significant issue to be a consideration in revising the BCR No. 2 of 2015 or in the nuclear licensing procedures.

In conducting the plant design review, the fulfillment of general and specific nuclear reactor design requirements carried out using BCR No. 1 of 2011. The reviewer compares all compliance of the article required in this BCR with information facility data declared in the last FSAR. In general, this BCR is sufficient to review the current conditions of general and specific design installation described in the review of the plant design above. Additionally, regarding the design of buildings and structures of the reactor, the reviewers also take into account the acceptance criteria stated in the BCR related to site evaluation of nuclear installation.

The actual condition of important SSCs to safety is reviewed by analyzing several data of inspection, maintenance, calibration, surveillance, functional tests, and operation reports for ten years. These data are interrelated to each other of the safety factors and able to give the necessary complement information. For instance, in reviewing plant design safety factors, such data can analyze whether SSCs are reliable and easy to operate, inspect, maintain, and test. In advance, this information and data include reactor parameter operation data collected can predict plant performance like availability factor, capacity factor, number of reactor trips, and unplanned shutdowns. Those data will be review in the safety performance of the safety factor.

The data collected in the PSR during 2005-2015 used as a temporary database system of operating installation data that describes the behavior of system and component operation, such as pressure, temperature, flow rate, conductivity, level of height, or humidity for important SSCs to safety. The databases associated with the event, deficiencies, anomalies, and deviation gathered to assist an essential view and analysis of operating experience from the failures of system or components, and maintenance deviation reports, which can transform into a database trending system representation. These systems will supply transparent data presentation that eases the diagnosis of monitored performance, and identify patterns, abnormal trends, recurrences, also quick plant management overview and action focus.

This trend analysis can provide information as an input on the aging safety factor review. For example, when the primary coolant pressure and temperature of the reactor are identified continuously outside the normal operating limits, the reviewer/inspector can rapidly recognize the possibility of degradation performance in those systems or components. Therefore, in the future, it is expected that BAPETEN creates a permanent database or integrating the inspection, maintenance, calibration, surveillance, functional tests, and operation reports data into Bapeten Licensing System and Inspection, as a useful additional tool to support BAPETEN supervision, to monitor reactor operation performance, as well as the licensing decision support system.

The second goal of the aging safety factor review is to know whether an effective management program of aging is suitable so that all required safety functions will be performing for the plant design lifetime or long-term operation. At the same time, additionally to PSR documents, the licensee also submits an aging study report in the operating license renewal that reflects the management implementation report of aging based on a predetermined program. The management program's effectiveness of the aging was decided by similitude the results of the aging management implementation as written in the study report with the management program composed by the licensee by referring to BCR No. 8 of 2008 on the Safety provision of non-power reactor aging management.

However, based on GR No. 2 of 2014, the aging management program is not included in the document for an operating license and is not approved by BAPETEN. Therefore, activities proposed in management programs that cover some process such as screening of SSCs, identification, and understanding of aging degradation mechanisms, determine critical of SSCs, and record of aging management data is potentially inappropriate due to the error of methodology with BCR No. 8 of 2008. As a result, the aging study report is potentially not accepted by the regulatory body. IAEA Specific Safety Requirement No. 3 shows that research reactor operating facility shall assure that the management program of aging performed effectively to maintain the aging of important SSCs to safety so that safety functions of SSCs are accomplished over the whole operating lifetime of the installation [21] as well in the IAEA SSG No. 12 [22]. This document declares that the operational program of licensee should have in place before and during operation. Such a program may be subject to approval by the regulatory body as appropriate is the management of aging. Therefore, as a lesson learned from the PSR review and assessment in the aging management aspect, BAPETEN should add an 'aging management program' in the revision of GR No. 2 of 2014 and should move the detailed requirements concerning the "aging management program" into a relevant BCRs.

CONCLUSION

PSR implementation in several regulating research reactor countries is varied. PSR regulations in Indonesia are sufficient. However, it needs updating specifically regarding the aging management program in GR No. 2 of 2014 and PSR formal procedure in the BCR or nuclear reactor licensing procedure. The results of PSR give benefits and lessons for the licensee and the regulatory body. BAPETEN gains benefits as follows: an input to update the regulation and procedures and develop databases for the licensing decision support system. Meanwhile, BATAN gains lessons as follows: to know actual plant safety, also to plan and to set work priorities for safety improvements.

REFERENCES

[1] Badan Tenaga Nuklir Nasional, Chapter I Introduction in *Safety Analysis Report of GA Siwabessy Reactor Rev. 11*, Jakarta, 2019.

[2] Badan Pengawas Tenaga Nuklir, Bapeten Chairman Regulations Number 1 of 2017 on Inspection in Utilization of Nuclear Power Oversight, Jakarta, 2017.

[3] A. Khakim and Geni Rina, “Safety Reevaluation of Indonesian MTR-Type Research Reactor,” Paper presented at the 18th IAEA IGORR Conference, 4-8 Dec 2017, Sydney.

[4] Government of Republic of Indonesia, Government Regulation No. 2 of 2014 on Licensing of Nuclear Reactor and Utilization of Nuclear Materials, Jakarta,2014.

[5] Nuclear Regulatory Commission of United States, Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors, United States, 2009.

[6] Nuclear Regulatory Commission of United States, Reg Guide 2.7 Preparation of Updated Final Safety Analysis Reports for Non-Power Production or Utilization Facilities, United States, 2019.

[7] Australian Radiation Protection and Nuclear Safety Agency, Regulatory Guide for Periodic Safety and Security Review of Facilities v.2.0, NSW, 2019.

[8] International Atomic Energy Agency, Report of the Integrated Safety Assessment of Research Reactors (INSARR) Mission to the High Flux Reactor (HFR), Petten, 2016.

[9] J. Couturier, D. Rive. “Periodic Safety Review Management for French Research Reactors, Technical Support Organization Approach, , Institut de Radioprotection et de Sûreté Nucléaire, Fontenay-aux-Roses, https://www-pub.iaea.org/MTCD/Publications/PDF/P1360_ICRR_2007_CD/Papers/J.%20Couturier.pdf, accessed on 12 Sep 2020.

[10] T. Nemec, “Experience with Safety Reviews of Slovenian Research Reactor by PSR and IAEA INSARR Missions and the Stress Tests for the Krško NPP,” Presented at 18th IAEA IGORR Conference, 4-8 Dec 2017, Sydney.
[11] H. Gerstenberg et all, “First Periodic Safety Review of the FRM II After 10 Years of Routine Operation,” Technical University of Munich, ZWE FRM II, 85748 Garching – Germany,” Paper presented at 18th IAEA IGORR Conference, 4-8 Dec 2017, Sydney.
[12] Jin-Won Shin, Seung-Gyu Do, Jae-Sam Han, and Choong Sung Lee, “Status of a Periodic Safety Review of HANARO,” Paper presented at 18th IAEA IGORR Conference, 4-8 Dec 2017, Sydney.
[13] Bhuse P. P, “Safety Enhancement of Dhruva Reactor through Periodic Safety Review,” Paper presented at 18th IAEA IGORR Conference, 4-8 Dec 2017, Sydney.
[14] International Atomic Energy Agency, Specific Safety Guide No. SSG-25, “Periodic Safety Review for Nuclear Power Plants,” Vienna, 2013.
[15] Government of Republic of Indonesia, Government Regulation Number 54 of 2012 on The Safety and Security of Nuclear Installation, Jakarta, 2012.
[16] Badan Pengawas Tenaga Nuklir, Bapeten Chairman Regulation Number 2 of 2015 on Safety Verification and Assessment of Non-Power Reactor, Jakarta, 2015.
[17] Badan Tenaga Nuklir Nasional, *Document of Periodic Safety Review for GA Siwabessy Reactor 2005-2015* (Document Number: 002.002/RN 00 03/RSG 4.2), Jakarta, 2019.
[18] Badan Pengawas Tenaga Nuklir, Bapeten Chairman Regulation Number 1 of 2011 on Safety Design of Non-Power Reactor, Jakarta, 2011.
[19] Badan Tenaga Nuklir Nasional, *Document of Site Evaluation Report for BATAN Experimental Power Reactor* (seismic aspect) (Document Number: LET-RDE.01/RN 01/SN Revision 4), Jakarta, 2016.
[20] Badan Pengawas Tenaga Nuklir, Bapeten Chairman Regulation Number 8 of 2008 on Safety Provision of Non-Power Reactor Aging Management, Jakarta, 2008.
[21] International Atomic Energy Agency, Specific Safety Requirements No. 3, “Safety of Research Reactors,”, Vienna, 2016.
[22] International Atomic Energy Agency, Specific Safety Guide No. SSG-12, “Licensing Process for Nuclear Installations,” Vienna, 2010.

Study of Application of Safety Culture in PTBBN Using Lime Survey Method

Mugiyono[1, a)], Nofriady Aziz[1,a)], Sri Wahyuningsih[1,a)], Nudia Barenzani[1,a)],

[1]*Pusat Teknologi Bahan Bakar Nuklir – BATAN*

Email: [a)]nofriady-aziz@batan.go.id

Abstract. Centre of Nuclear Fuel Technology (PTBBN) conducts a self-assessment to obtain information on the application of safety culture in developing programs to implement safety culture in a timely and appropriate manner. The study on the application of safety culture has been carried out to determine the level of development of safety culture in research activities at PTBBN which is carried out by conducting a self-assessment of PTBBN safety culture, knowing weak attributes and strong attributes in the assessment of safety culture self assessment, and to determine a safety culture strengthening program. The study method is by filling out the questionnaire using a lime survey application that contains as many as 39 statement attributes to be assessed with a score of 0 to 5. The revelation material is based on BATAN Regulation No. 4 of 2019 which is divided into 5 characteristics of safety culture. Self Assess Results PTBBN on 2019 obtained a value that is equal to 810.07 with category B (in the assessment with 5 attribute scales). In the analysis of the results there are 5 weak attributes, there are 17.89% - 25.26% of employees who disagree, namely at and 5 strong attributes, there are 85.79% - 93.16% of employees who agree with the statement based on BATAN Regulations No. 4 of 2019 and IAEA GSG 5.3. A safety culture strengthening program to strengthen weak attributes and improve strong attributes. Key words : Safety Culture, Lime Survey, PTBBN

INTRODUCTION

Centre of Nuclear Fuel Technology of the National Nuclear Energy Agency (PTBBN BATAN) in the implementation of the Occupational Safety and Health Management System since 2013 has been certified in accordance with the Standard BATAN SB-006-OHSAS 18001 - 2008 issued by PSMN BATAN. This certification is evidence of the commitment of the leaders of PTBBN to make safety a top priority in every activity by implementing safety culture in accordance with BATAN Regulation No. 4 of 2019. Through Safety Culture, the nature and behavior of individuals and organizations are able to give priority and primary considerations for safety and can be seen clearly and observed both in organizations and individuals through all activities at all levels in PTBBN.[1] Safety culture is recognized as a very important component in the performance of nuclear safety.[2] Safety Culture is also defined as an environment that supports reporting, does not blame each other, involves top-level leadership and is focused on the system.[3] The management system that has been established must be able to support individuals in carrying out their duties related to safety, security, health and quality in an integrated manner, taking into account interactions between individuals, technology and organizations. Safety management systems have a broader role as a complement to the framework from planning, controlling and monitoring activities related to nuclear plant safety, radiation safety and the environment.[1] To be able to develop programs to implement safety culture in a timely and appropriate manner, PTBBN conducts a self-assessment to obtain information or an initial portrait of the application of safety culture. The results of this self-assessment are used to determine the level of safety culture to be achieved.

The existence of safety culture in the workplace can be seen from the mindset, attitude patterns and patterns of action of personnel regarding matters related to safety, especially in handling radioactive materials that have the potential for radiation hazards such as in PTBBN.[4] High safety will be achieved if the personnel in the plant adopt a safety culture. o the extent that the management of the PTBBN organization enhances the application of safety culture in the workplace, especially in the management of operational activities for research, it is necessary to carry out continuous self-assessments by re-surveying the implementation of safety culture. The safety culture self-assessment at PTBBN was carried out using the lime survey application questionnaire method of 190 employees.

Lime survey is a sophisticated online survey system for creating quality online surveys. Lime Survey is an open source online survey application written in PHP with a MySQL database that is centered on 1 server, namely in PPIKSN BATAN.[12] Lime Survey is designed to be easy to use, allowing users to develop and publish surveys, and collect responses, without doing any coding.[13] The lime survey application has the advantage of being able

ISSN: 1412-3258 © 2020 BAPETEN – FMIPA UI

to create an unlimited number of surveys, followed by respondents in an unlimited number, ready-made questions that can be imported, import and export data to text, CSV, PDF, SPSS, R, queXML and MS formats Excel.

The results of the survey can be used by management to continue to develop a culture of safety that high safety can be achieved.[5] Implementation of lime survey application has been used in SMK Cengkareng 2 based on open source with virtualization technology. Currently the safety culture questionnaire filling in BATAN conducted with lime application survey.[13] The purpose of conducting a safety culture self-assessment is to conduct a PTBBN safety culture self-assessment, identify weak and strong attributes in the safety culture self-assessment, and to determine a safety culture strengthening program. The assessment is done by comparing the conditions of safety culture that runs in the organization when the assessment is carried out with parameters 5 (five) characteristics and 37 (thirty-seven) safety culture attributes. Each organization has diverse perceptions of the concepts of safety culture, and needs positive action to increase that understanding.[4] Terminology Safety Culture or Safety Culture was first used after the accident at the Chernobyl reactor on April 26, 1986. The International Nuclear Safety Advisory Group-4 (INSAG-4) defines: "Safety culture that is compatible with the differences contained in relationships and relationships that occurs in organizations and individuals that are requested commensurate with their interests ".[6] The above statement implies that safety culture must be a priority at nuclear power centers.[7] According to BATAN Regulation No. 4 of 2019, Safety Culture is a characteristic and attitude towards organizations and individuals that determine safety as a top priority. Nuclear facilities are facilities, along with supporting facilities, namely land, buildings and equipment where radioactive materials and substances are produced, processed, used, or stored in quantities whose safety needs to be considered.[8]

Safety culture in the workplace is closely related to the nature, attitudes and behavior of survivors in each part of the organization and individual workers.[9] Every individual in the organization has a different level of understanding and perception of the concept of safety culture. It is therefore necessary activities promotive, curative and persuasive so that each individual has the same understanding and perception of the concept of safety culture in the organization. The basic principle that needs to be understood and understood by every individual is that in the concept of safety culture all obligations relating to safety must be carried out correctly, thoroughly, can be accounted for and given the highest priority.

Safety culture in BATAN is a reflection of the values, attitudes and behaviors that are held and owned by every individual, both Policy Makers, High Leaders, Managerial / Structural Officers, Functional Officers and Implementers. Safety culture is based on the belief that safety is important and primary and becomes a shared responsibility. Those values are a basic framework, directions and goals for each individual in carrying out the duties and responsibilities of each. Implementation of safety culture is strongly influenced by the organization and activities with one another therein may affect each other and together affect the performance of the organization of the safety and even the overall performance of the organization.[10]

The characteristics of safety culture as a strategy for developing safety culture include structured attitudes and behavior. The characteristics of safety culture are a series of interaction processes of each individual involved in contributing to achieving high safety performance. Safety culture consists of five (5) characteristics as shown in **FIGURE 1** and described to 37 (thirty-seven) attributes of safety culture.[8]

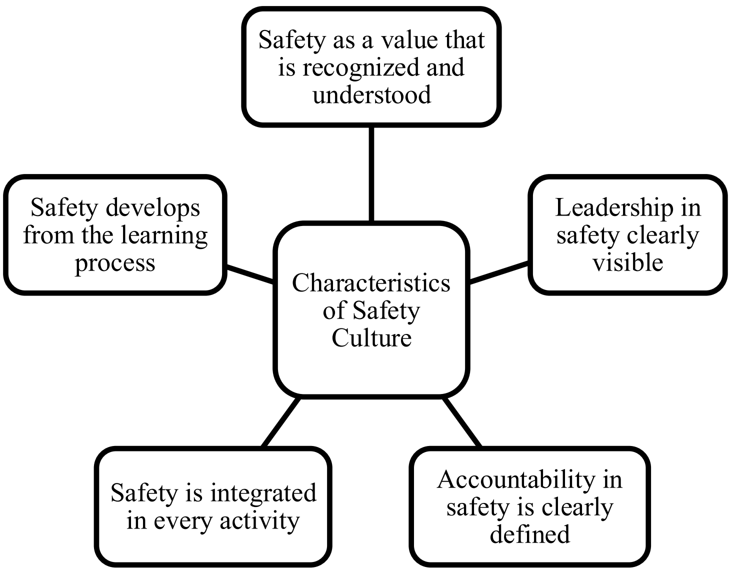


FIGURE 1. Characteristics of Safety Culture

1. Safety as a value that is recognized and understood. Organizations that have safety characteristics as recognized and understood values can be seen from the commitment of organizational leaders who place safety as the first priority reflected in documentation, communication, and decision making. Safety is the main consideration for organizational leaders in allocating resources, where the resources in question are not only related to the budget, but also ensures the availability of sufficient numbers of employees with adequate competence.
2. Leadership in safety clearly visible. Leadership in safety is reflected in the attitude of organizational leaders who prioritize safety, and this attitude is clearly seen by colleagues and employees below it both in daily communication and in regular meetings where safety issues are discussed. Organizational leaders understand their duties and responsibilities well and are able to set an example. 'The example demonstrated by the Leader will encourage the improvement of safety culture in the organization, for example by visiting employee facilities and workspaces regularly to assess safety performance, as well as providing space for employees to get involved and play an active role in enhancing safety culture.
3. Accountability in safety is clearly defined. From a safety standpoint, individual roles and responsibilities in terms of safety must be clearly defined. roles and responsibilities are outlined in the form of detailed job descriptions. Accountability in safety means that the work objectives have been determined, targets have been set, then an evaluation of the progress of the activity is carried out so that it can be assessed whether the safety targets have been achieved. Accountability in safety is reflected in the relationship between the organization and the regulatory body and other external parties. A good relationship is the existence of open communication, trust, discussion and consultation with each other so that each party can carry out their duties and responsibilities and formulate more effective future work plans.
4. Safety is integrated in every activity. Safety considerations carried out by leaders and employees in the organization must consider all aspects including environmental and industrial safety, as well as security. The integration of all aspects of safety appears in the entire work process starting from planning, implementation, and evaluation. All processes are carried out by prioritizing quality. 'Documentation and procedures can be understood easily, can be accessed by all employees, and can be guaranteed enforceability.
5. Safety develops from the learning process. The organization must have the philosophy that every problem is an opportunity to get learning. Every individual has the willingness to share experiences with each other in terms of safety. At the organizational level, the learning process can be done through training, comparative studies, workshops to share experiences, and other forms of activities that can enhance safety culture.

Safety culture self-assessments are shown in **TABLE 1**.

Table 1. Safety Culture Assessment Scheme

Score	Alphabet	Note
840 – 1000	A	Very Good
680 – 839	B	Good
520 – 679	C	Enough
360 – 519	*D*	*Less*
200 – 359	E	Bad

Self-assessment can be done through several methods, namely document reviews, questionnaires, observations, focus groups and interviews. [11] However, none of these methods can simultaneously measure all elements of the intangible safety culture, such as norms, values, beliefs, attitudes or behaviors. Each method has strengths and weaknesses in measuring these cultural elements. Therefore, it is highly recommended to use the five methods together, although the survey method using a questionnaire can be used as a baseline measurement. With the survey method using a questionnaire, assessors can obtain information that represents the whole or part of an organization. The results of filling out the questionnaire can be quantified and can be compared between groups and from time to time. Fill out the questionnaire using the lime survey link.

METHODOLOGY

The method of conducting self-assessment in PTBBN is carried out for all employees in PTBBN and conducted on December 2 - December 20, 2020. This self-assessment is carried out by filling out the questionnaire in accordance with BATAN Regulation No. 4 of 2019 with reference to 5 characteristics and or 39 attributes of safety culture. This self-assessment was carried out online and filled up to 190 people. Each employee can fill in via email each employee or can access using the website http://223.25.97.91:8006/survey/index.php/265794?token=(token number) in the browser. Each employee will get a different token number. Each question can be scored between 0 to 5. The score is divided into five levels, namely: 1: the category of disagree, 2: the category of disagree, 3: the category of quite agree, 4: the category of agree and 5: the category of strongly agree. Instructions on filling out the questionnaire are given clearly so as to facilitate personnel in answering these questions. The lime survey display can be shown in **FIGURE 2**.

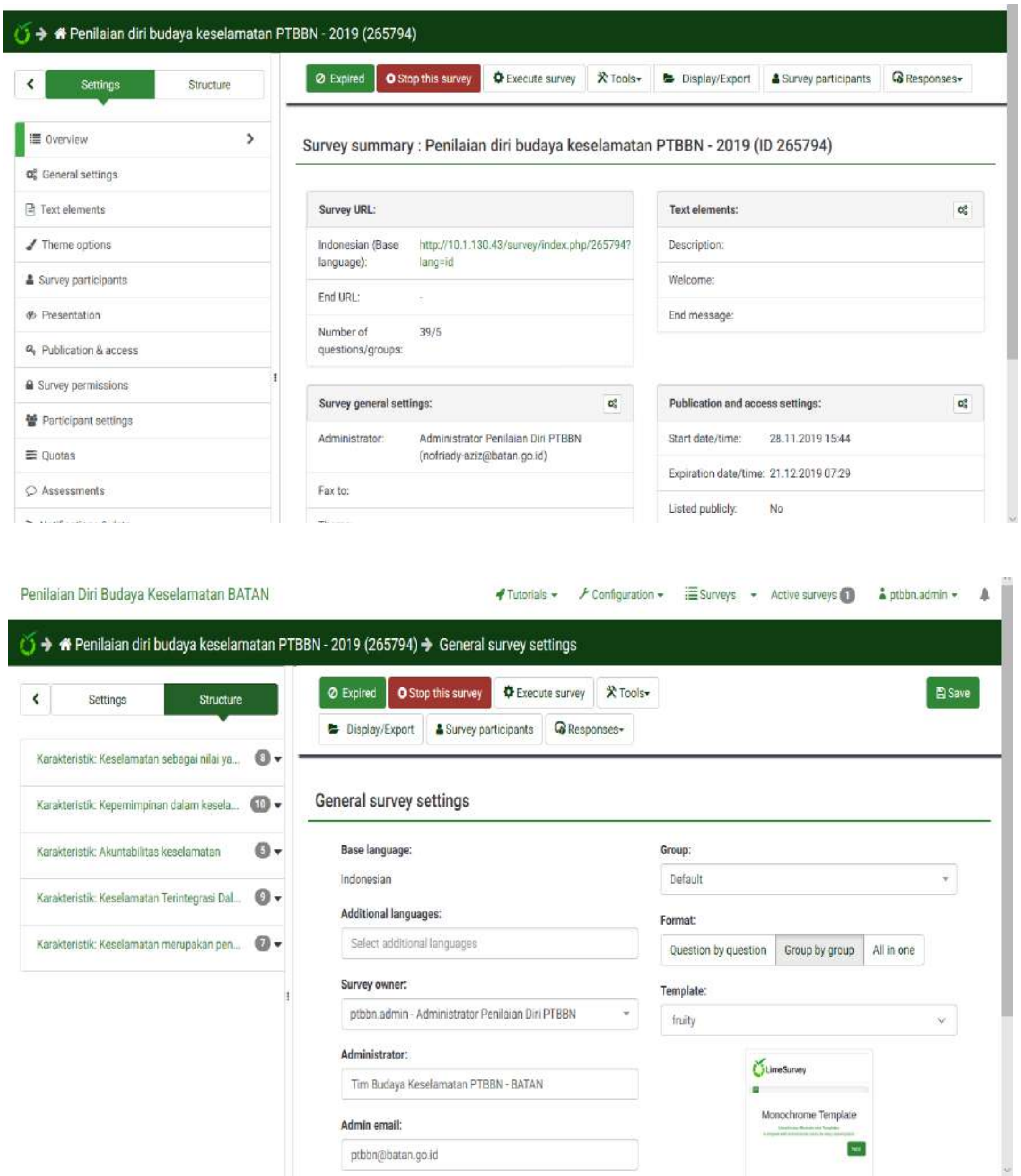


FIGURE 2. *Lime Survey*

RESULT AND DISCUSSION

Attribute Safety Culture Assessment forms (questionnaires) that have signed as many as 190 people. The results of the questionnaire assessment are based on surveys from data collection through questionnaires which are then analyzed based on the weighting stated by ranking the results of the analysis for each attribute or characteristic in accordance with Regulation 4 of BATAN No. 4 of 2019. The results of the self-assessment carried out are outlined in Table 2 of PTBBN self-assessment results 2019 using the lime survey method.

TABLE 2. Self-Assessment PTBBN 2019

Characteristics of Safety Culture	Mark	Convert mark to scale 1000	Category	Rank Category
Safety as a value that is recognized and understood	4,11	821,18	B	Good
Leadership in safety clearly visible	4,05	809,26	B	Good
Accountability in safety is clearly defined	4,03	806,95	B	Good
Safety is integrated in every activity	3,98	795,79	B	Good
Safety develops from the learning process	4,09	817,14	B	Good
Score of Safety Culture Survey	4,05	**810,07**	B	Good

Based on the results of the 2019 PTBBN Self-Study shown in **TABLE 2** that the work unit safety culture survey value was 810.07 with category B (in the assessment with 5 attribute scales) according to the safety culture assessment scheme in **TABLE 1** namely 680 - 839 of 37 attributes and 5 characteristics. The results of the self-assessment were obtained from the import of lime survey results into the Microsoft Excel program. Safety as a value that is recognized and understood get a value of 4.11 (number 4 states agree) so as to obtain category B. Leadership in safety clearly visible get a value of 4.05 (number 4 states to agree) so as to obtain category B. Accountability in safety is clearly defined get a value of 4.03 (number 4 states agree) so as to get category B. Safety is integrated in every activity get a value of 3.98 (number 4 states agree) so as to obtain category B. Safety develops from the learning process get a value of 4.09 (number 4 states agree) so as to obtain category B. The average value of the 5 attributes of 4.05 was converted to a scale of 1000 according to the calculation of safety culture assessment values according to BATAN Regulation No. 4 of 2019 and 810.07 obtained figures. According to the safety culture assessment scheme in **TABLE 1** namely 680 - 839 for the B value category (good).

TABLE 3. Weak Attribute

Attribut e No	Characteristic	Attribute	Percenta ge (%)
7	Safety as a value that is recognized and understood	A proactive approach to long-term safety issues in my work unit is part of the decision making process	17,89
9	Leadership in safety clearly visible	Structural officers in my work unit have the ability to resolve conflicts	25,26
4	Accountability in safety is clearly defined	Structural officials in my work unit delegate responsibility to employees with the proper authorities	23,16
8	Safety is integrated in every activity	Cooperation between the field and expertise has been running well in my work unit	25,26
1	Safety develops from the learning process	The attitude of asking questions has been developed in every individual in my work unit	18,42

Based on the results of the 2019 PTBBN safety culture self-assessment in Table 3, it appears that there are 5 weak attributes. These attributes are weak because the 5 attributes get the lowest response among other attributes, there are 17.89% - 25.26% of employees who disagree with the statement based on BATAN Regulation No. 4 of 2019 and IAEA GSG 5.3.[8] [14] These attributes include attribute number 7 Safety as a value that is recognized and understood, so of the 190 employees who filled out there were 34 employees who stated disagreement. This is made possible by an understanding of the safety values related to attribute number 7 is still unclear. Therefore, there needs to be socialization related to these attributes. Attribute number 9, Leadership in safety clearly visible, there are 48 employees out of 190 employees who rate not agree. PTBBN has carried out efforts to improve the competence of structural officials since 2018 in collaboration with anonymous consulting and this will still be followed up on. Attribute number 4 for accountability in safety defined is 44 employees who disagree. Follow-up is the same as attribute number 9 points b. Attribute number 8 in safety is integrated in every activity that there are still 48 employees who rate it low, so this will be followed up with gathering, joint gymnastics, coffee morning and other togetherness events in order to increase openness of communication and cooperation. Attribute number 1 safety develops from the learning process there are still 35 employees who underestimate the occupational safety and health task force will seek to improve communication through socialization and coaching.

TABLE 4. Strong attribute

Attribute No	Characteristic	Attribute	Percentage (%)
5	Safety as a value that is recognized and understood	Safety related matters have been considered in my work unit's activity plan	91,58
1	Leadership in safety clearly visible	The head of my work unit shows a clear commitment to safety	89,47
1	Accountability in safety is clearly defined	My work unit fosters good relations with supervisory units / institutions	90,00
3	Safety is integrated in every activity	Documentation and procedure (SOP) in my work unit is in conformity with the requirements	85,79
4	Safety develops from the learning process	My work unit uses safety related experience for the learning process	93,16

Based on the results of the 2019 PTBBN safety culture self-assessment in **TABLE 4**, it appears that there are 5 strong attributes. This attribute is strong because the 5 attributes get the highest response among other attributes, there are 85.79% - 93.16% of employees who agree with the statement based on BATAN Regulation No. 4 of 2019 and IAEA GSG 5.3.[8] [14] The attribute includes attribute number 5 Safety as a value that is recognized and understood, so of the 190 employees who filled in there were 174 employees who agreed. This is because Safety related matters have been considered in my work unit's activity plan such as having a briefing every morning before the activity. Attribute leadership in safety clearly visible that there are 170 employees out of 190 employees who rated agree. The head of my work unit shows a clear commitment to safety. Attribute number 1 for accountability in safety, 171 employees agreed. This is due to a good relationship with supervisory agencies such as BAPETEN. Attribute number 3 Safety is integrated in every activity that there are still 163 employees who highly rate. Documentation and procedure (SOP) in my work unit is in conformity with the requirements. Attribute number 4 safety develops from the learning process that there are still 177 employees who rate it highly, because my work unit uses safety related experience for the learning process.

The safety culture strengthening program can be implemented can be seen in **TABLE 5**. The safety culture strengthening program is a generic program created by BATAN based on sharing the implementation of the safety culture strengthening program in each work unit. The program is a recommendation of the BATAN 2019 safety culture team to be implemented in the work unit. This program was carried out in Yogyakarta PSTA BATAN at the 2019 K3 month workshop.

TABLE 5. Safety Culture Strengthening Program

No	Safety Culture Strengthening Program	Application on Weak Attributes
1	Hazard identification and risk assessment activities at EFEI and RMI	Attribute 1 Characteristics 5 Attribute 8 Characteristics 4
2	Socialization and internalization of the importance of safety culture through various media	Attribute 7 Characteristics 1 Attribute 8 Characteristics 4
3	First Aid and Resque Workshop	Attribute 4 Characteristics 3 Attribute 8 Characteristics 4
4	Fire fighting and fire risk assessment training	Attribute 4 Characteristics 3 Attribute 8 Characteristics 4
5	Behavior based safety is an intervention program for unsaved actions	Attribute 9 Characteristics 2 Attribute 1 Characteristics 5
6	Campaign for the proper use of PPE through poster	Attribute 7 Characteristics 1 Attribute 8 Characteristics 4
7	Implementation of 5R activities (concise, neat, clean, caring and diligent) once a month	Attribute 4 Characteristics 3 Attribute 8 Characteristics 4
8	Brifieng morning every day before entering the laboratory	Attribute 4 Characteristics 3 Attribute 8 Characteristics 4
9	Workshop safety leadership	Attribute 9 Characteristics 2 Attribute 4 Characteristics 3
10	Comparative study of the implementation of safety culture	Attribute 7 Characteristics 1 Attribute 1 Characteristics 5
11	Safety culture self-assessment	Attribute 7 Characteristics 1 Attribute 9 Characteristics 2 Attribute 8 Characteristics 4

CONCLUSION

The results of the 2019 PTBBN Self-Assessment obtained a value of 810.07 with category B (on an assessment with 5 attribute scales). In the analysis of the results there are 5 weak attributes, there are 17.89% - 25.26% of employees who disagree, namely at and 5 strong attributes, there are 85.79% - 93.16% of employees who agree with the statement based on BATAN Regulations No. 4 of 2019 and IAEA GSG 5.3. A safety culture strengthening program to strengthen weak attributes and improve strong attributes.

ACKNOWLEDGMENTS

Authors say thanks to PTBBN and those who have assisted in this research activity.

REFERENCES

[1] Supriyanto.S, dkk., 2013. *Kajian Penerapan Budaya Keselamatan Di Reaktor Ga. Siwabessy*. Serpong. Prosiding Seminar Nasional Teknologi Dan Aplikasi Reaktor Nuklir ISBN 978-979-17109-8-5

[2] E. Lestariningsih, dkk., 2017, *Evaluasi Hasil Penilaian Diri Budaya Keselamatan PSTA*. Yogyakarta. Prosiding Pertemuan Dan Presentasi Ilmiah Penelitian Dasar Ilmu Pengetahuan Dan Teknologi Nuklir Issn 0216-3128.

[3] AORN Journal. 2006. Aorn Guidance Statement: Creating A Patient Safety Culture. Aorn Journal, 83, 936-942.

[4] Nudia Barenzani, 2009. *Kajian Penerapan Budaya Keselamatan di IEBE*. Serpong. Hasil-Hasil Penelitian Esn Tahun 2009 Issn 0854 – 5561.

[5] Titus Permadi Setiawan, 2012. Survei Online Penunjang Penelitian Praktis Dan Akademis. Semarang. Seminar Nasional Teknologi Informasi & Komunikasi Terapan 2012 (Semantik 2012) ISBN 979-26-0255–0.

[6] International Atomic Energy Agency. 1991. Safety Culture: a Report by the International Nuclear Safety Advisory Group (Safety Series No. 75 INSAG-4). Vienna.IAEA.

[7] Heni Na, dkk., Materi Rekualifikasi Petugas Proteksi Radiasi Evaluasi Penerapan Budaya Keselamatan Instalasi Nuklir, Jakarta. BAPETEN.

[8] Peraturan Batan No 4 Tahun 2019 tentang Penerapan Budaya KeselaMatan Di Lingkungan Badan Tenaga Nuklir Nasional

[9] Purwaningsih. R, dkk., 2019. Penilaian Budaya Keselamatan Dengan Metode Scart (Safety Culture Assessment Review Team) Pada Badan Pengelola Instalasi Nuklir). Semarang. Jurnal Teknik Industri, Vol. 14, No. 1.

[10] Situmorang. 2013. Evaluasi Budaya Keselamatan Untuk Prioritisasi Pentingnya Characteristics/Attribute Pada Instalasi Nuklir Dengan Teknik Ahp (Analytic Hierarchy Process). Jurnal Teknologi Pengelolaan Limbah, Vol 16 No 1 Juli 2013, Pusat teknologi limbah radioaktif. BATAN, Jakarta

[11] Safety Report Series, 2016. No 83, *Performing Safety Culture Self-Assessments*, IAEA.

[12] Limesurvey Project Team / Carsten Schmitz (2012). / Limesurvey: An Open Source Survey Tool /Limesurvey Project Hamburg, Germany.

[13] Ahmad Miftahudin. 2014. Implementasi Aplikasi Limesurvey Di Smk Cengkareng 2 Berbasis Open Source Dengan Teknologi Vitrualisasi. Jakarta. Universitas Mercu Buana

[14] IAEA GSG 5.3

[15] Allen, P. J., & Roberts, L. D. 2010. *The Ethics of Outsourcing Online Survey Research*. International Journal of Technoethics, 1(3), 35-48

[16] Wijayanto, A. dkk., 2014. *Penyediaan Sistem Informasi Budaya Keselamatan PTLR – BATAN*. Yogyakarta. Seminar Nasional X SDM Teknologi Nuklir.

Conversion of ORIGEN2 Output for MCNP6 Calculation

Fibra Rhoma Firmanda[1, a)]

[1]BAPETEN-P2STPIBN

[a)]f.rhoma@bapeten.go.id

Abstract. Reliable and accurate neutronic calculations are required to perform deterministic safety analyzes in nuclear facilities. The Continuous-energy Monte Carlo method using MCNP6 software is one of the reliable neutronic calculation methods. However, MCNP6 requires fissile material data to perform neutronic calculations, especially if burn-up has been carried out and the fuel is no longer fresh. MCNP6 needs to be coupled with other software that calculates the build-up, decay, and processing of radioactive materials to produce fissile material data. One such software is ORIGEN2. The computer code develops in this paper connects ORIGEN2 and MCNP6 by providing converted and filtered fissile material data. This computer code uses the array manipulation method with the Python programming language in its manufacture. Based on the comparison of the initial data and the processed data, it shows accurate results. Then the verification results by calculating the keff in the 2000 Triga Reactor code using the converted fissile material data also showed the appropriate results. This computer code can be used to convert fissile material data from ORIGEN2 to MCNP6 and improve the accuracy of neutronic calculations.

Keywords: ORIGEN2, MCNP6, computer code

INTRODUCTION

Sufficient computer code can be used to perform deterministic safety analyzes in nuclear facilities. Deterministic safety analysis requires reliable and accurate neutronic calculations. One of the reliable methods is the Continuous-energy Monte Carlo method [1] because of the accuracy of its geometric modeling and physical phenomena.

MCNP is a 3-dimensional simulation software that uses the Continuous-energy Monte Carlo method and can be used for the calculation of multiplication factors, reaction rates, neutron flux, spectra, etc. [2]. One of the advantages of MCNP is that it can be run in parallel using MPI (Message Passing Interface) [3] and is easy to use to perform complex geometry calculations compared to conventional neutron transport deterministic computer codes.

MCNP requires data on fissile product material and geometry to perform neutronic calculations, especially if the fuel has gone through the burn-up process and is no longer fresh fuel. MCNP needs to be coupled with a computer code capable of calculating the accumulation, decay, and processing of radioactive materials to complement the data on the fissile product material. The MCNP itself is capable of calculating the accumulation, decay, and processing of radioactive materials but the process takes a long time. The computer code that is widely used to calculate the accumulation, decay, and processing of radioactive materials and obtain material data on fissile products is Origen [4]. Some computer codes for fuel fraction calculations have been developed including MCORE [5] connecting MCNP4C and ORIGEN2.1 using the "modified predictor-corrector" approach, then MCWO [6] connecting MCNP and ORIGEN2 using LINUX BASH script file.

This study aims to develop a program that converts ORIGEN2 output for MCNP6 calculation using the Python programming language. Later, hundreds or thousands of data will be collected, filtered, and converted from ORIGEN2 to be processed in MCNP6.

METHODOLOGY

This computer code was developed using NumPy function to manipulate array with Python programming language. The computer code that develops in this paper is divided into 4 parts like the diagram in Figure 1. The input and output of this program are in the form of text files.

The first part is used to import data into an array that has 2 columns. The import method of this data is using the Numpy function from the Python programming language. The first column contains the atomic number and mass number encoded into the Nuclide Identification Number, while the second column contains the atomic gram value of the material. Data containing in these 2 columns are stored in an array for later processing. In addition to importing data to be processed. This computer code also imports the MCNP6 library. This library will be used later in the third section for filtering. Imported libraries are also stored as arrays.

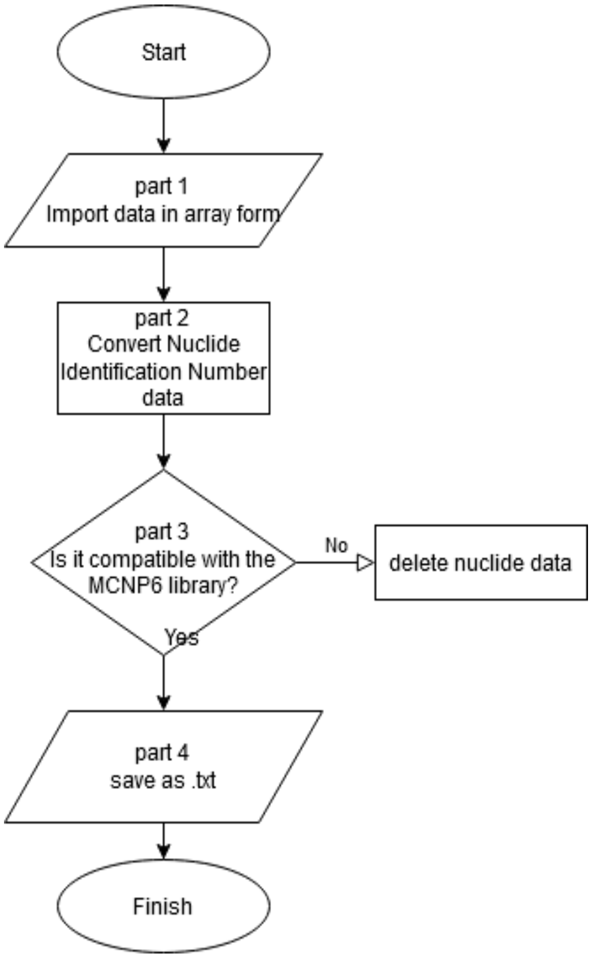


FIGURE 1. Computer Code Flow Diagram

The second part is changing the Nuclide Identification Number from Origen2 to MCNP6. The data processed in this section is the first column of the array that was created before. The comparison of the Nuclide Identification Number format for MCNP and ORIGEN is as follows:

The nuclide identifier of ORIGEN2 is defined by six-digit nuclide identifiers as

NUCLID = 10 000*Z+10*A+M

Where

NUCLID = six-digit nuclide identifier
Z = atomic number of the nuclide
A = atomic mass of the nuclide
M = state indicator, 0 = ground state, 1 = excited state

The nuclide identification number of MCNP with the form ZZZAAA.nnX, where

ZZZ = atomic number
AAA = mass number (000 for elements)
Nn = unique table identification number
X = C for continuous-energy neutron tables
X = D for discrete-reaction tables

For example, the nuclide identifier of ORIGEN2 for Uranium 235 is

= 10 000*Z+10*A+M
= 10 000*92+10*235+0
= 920 000+2350+0

= 922350

And example the nuclide identifier of MCNP for Uranium 235 is

	= ZZZAAA.nnX
ZZZ	= 92
AAA	= 235
nn	= 80
X	= C
	= 92235.80C

Nuclide Identification Number in MCNP and ORIGEN have differences, especially in the last digit where the last digit MCNP represents the type of energy while in ORIGEN represents the condition of the excited atom or not, the default condition of the last digit in this computer code for ORIGEN2 is 0 while MCNP is 80C.

The third part is to filter the Nuclide Identification Number that is not in the MCNP6 library. ORIGEN2 and MCNP6 have different Nuclide Identification Number libraries. Some data in the ORIGEN2 library is not in the MCNP6 library, this can cause an error when running input made from this computer code on MCNP6. For this reason, it is necessary to filter data from ORIGEN2 which is incompatible with the MCNP6 library. Filtering is done by taking the data library array MCNP6 that was created in the first section and then comparing it automatically one by one using the logic "if" in the phyton programming language so if there is data that is not appropriate will be deleted.

The fourth part is exporting the array data to a .txt format file. The exported data is in the form of 3 pairs of columns containing the Nuclide Identification Number and atomic gram of the substance.

MCNP6

Monte Carlo N-Particle (MCNP) is a general-purpose, three-dimensional simulation software that transports 37 different types of particles for criticality calculations, shielding, dosimetry, response detectors, and many other applications. Monte Carlo itself is a statistical calculation algorithm that replicates random numbers to solve complex problems that cannot be solved analytically. This software was developed starting in the 1940s by the Los Alamos National Laboratory (LANL) which investigated the problem of transporting neutrons on first-generation computers. This device began to develop rapidly since the 1960s due to the development of computer technology to come out with its latest product, MCNP version 6 [2].

ORIGEN2

Origen2.1 is a computer code for radioactive depletion and decay calculations developed by the Oak Ridge National Laboratory (ORNL). ORIGEN is able to perform neutron calculations that give a wide variety of nuclear material (buildup, decay, and processing of radioactive material) in an easily applicable form.

In principle, ORIGEN2.1 is used to calculate the radionuclide composition and other related properties of nuclear materials. Materials commonly characterized are spent fuel, radioactive waste (usually high level waste), recovered elements (e.g., uranium, plutonium), uranium ore and mill tailings, and gaseous effluent streams (e.g., noble gases).

Phyton Programming Language

Python is a high-level programming language that uses for multi-objective and has broad used. This programming language is designed to be easy to read and simple to write. One of the advantages of Phyton is it has fewer lines than the other programming languages such as C/ C++.

In the computer code developed in this paper, the NumPy function is used to process data. NumPy (Numerical Python) is a Python library that focuses on scientific computing. NumPy has the ability to construct N-dimensional array objects, which are similar to lists in Python. The advantages of NumPy arrays compared to lists in Python are less memory consumption and faster runtime. NumPy also makes it easier to use Linear Algebra, especially operations on Vector (1-day array) and Matrix (2-day array). The basis of NumPy features is vectorization. Vectorized code has many advantages, among which are: vectorized code is more simple and readable, fewer bugs because have fewer lines than the other function, the code has similarity with standard mathematical notation and more efficient and easy to manage.

RESULTS AND DISCUSSION

The results obtained in this program are in the form of a text file containing the composition of the nuclide and atomic grams. These results are compared with preliminary data and are seen as being compatible with the original purpose of this computer code. Figure 2 shows the input.

```
1  10010 3.58442535E+01  10020  6.65894663E-03  10030 3.42760986E-10 0 0.0
1  20030 6.69065335E-13  20040  1.07086557E-19 400900 1.14294701E+01 0 0.0
1 400910 2.49410081E+00 400920  3.80041027E+00 400930 1.06370455E-04 0 0.0
1 400940 3.86462712E+00 400950  2.02624888E-05 400960 6.21896684E-01 0 0.0
1 400970 4.18951636E-12 410930  8.65757932E-14 410931 1.61578335E-12 0 0.0
1 410940 2.99298761E-19 410950  2.66821917E-06 410951 7.65425412E-09 0 0.0
1 410960 1.06742194E-14 410970  3.20996159E-13 410971 3.91366195E-15 0 0.0
1 420940 3.30283242E-25 420950  3.63174394E-07 420960 6.95526067E-11 0 0.0
1 420970 1.16501656E-06 420980  6.05196518E-11 420990 1.05170109E-17 0 0.0
1 430990 2.74166102E-17 440990  2.19538609E-24 441000 1.10769743E-21 0 0.0
1 441010 1.41043634E-25      0  0.00000000E+00      0 0.00000000E+00 0 0.0
2  20040 1.54154411E-10 812080  6.07942658E-25 822070 1.68417528E-23 0 0.0
2 822080 2.27705758E-21 822110  2.12100736E-25 822120 3.51815177E-22 0 0.0
2 832120 3.33719181E-23 862200  5.10729011E-25 882230 9.67379118E-23 0 0.0
2 882240 2.90454153E-21 882280  1.49409664E-24 892270 2.27506287E-18 0 0.0
2 902270 7.88851818E-22 902280  9.67899027E-19 902290 3.84763246E-23 0 0.0
2 902300 2.65629828E-20 902310  6.19688328E-13 902320 1.72625737E-12 0 0.0
2 902340 3.16377350E-12 912310  5.68288212E-12 912320 5.12497367E-18 0 0.0
2 912330 2.07444764E-15 912341  1.06675250E-16 912340 4.76482800E-17 0 0.0
2 922320 3.50751918E-15 922330  2.72939930E-16 922340 7.08810343E-13 0 0.0
```

FIGURE 2. Example of input

Seen in Figure 2, there are three pairs of columns containing the value of the nuclide identifier and the gram value of the nuclide atom. When compared in Figure 3 the resulting output shows that the format of the nuclide identifier has changed and filtering also occurs so that some of the nuclide data is removed according to the MCNP6 data library.

```
1001 3.58442535E+01 66160 6.97227996E-12 66163 9.98951810E-10
2003 6.69065335E-13 66162 2.85787394E-09 68166 5.72423255E-11
40091 2.49410081E+00 67165 1.69116374E-10 69170 6.95253221E-21
40094 3.86462712E+00 68168 1.19220486E-12 1003 3.42760986E-10
41094 2.99298761E-19 68170 6.88309588E-25 40090 1.14294701E+01
42094 3.30283242E-25 1002 6.65894663E-03 40093 1.06370455E-04
42097 1.16501656E-06 2004 1.07086557E-19 40096 6.21896684E-01
43099 2.74166102E-17 40092 3.80041027E+00 41093 1.61578335E-12
44101 1.41043634E-25 40095 2.02624888E-05 41095 7.65425412E-09
2004 1.54154411E-10 41093 8.65757932E-14 42096 6.95526067E-11
82208 2.27705758E-21 41095 2.66821917E-06 42099 1.05170109E-17
88224 2.90454153E-21 42095 3.63174394E-07 44100 1.10769743E-21
90227 7.88851818E-22 42098 6.05196518E-11 82207 1.68417528E-23
90230 2.65629828E-20 44099 2.19538609E-24 88223 9.67379118E-23
90234 3.16377350E-12 90228 9.67899027E-19 89227 2.27506287E-18
91233 2.07444764E-15 90231 6.19688328E-13 90229 3.84763246E-23
92232 3.50751918E-15 91231 5.68288212E-12 90232 1.72625737E-12
92235 1.49759144E-01 92233 2.72939930E-16 91232 5.12497367E-18
92238 6.46353424E-01 92236 1.64911279E-03 92234 7.08810343E-13
93238 2.67139349E-12 92240 1.51030008E-15 92237 1.27285816E-07
```

FIGURE 3. Example of Output

From the above data, it can be seen that the computer code output is in accordance with the initial data provided. And filtering occurs to the Nuclide Identification Number data.

Verification of the results obtained from the computer code developed in this paper is done by calculating the keff of the Triga 2000 Reactor code using MCNP6. The first calculation was carried out without using the fissile product material produced by ORIGEN2. The second calculation is performed using the fissile product material produced by ORIGEN2 and has been converted using the computer code developed in this paper. The calculation results show that the keff generated from the first calculation is 1.048 greater than the keff in the second calculation, which is 1.01094. The fissile product material has a cross-section that absorbs neutrons resulting in a smaller keff in the second calculation. The accuracy of the neutronic calculations can be improved by involving the fissile product material data in the calculation.

CONCLUSIONS

Based on the comparison of the output generated by the computer code developed in this paper and the output produced by ORIGEN2, it shows that the conversion was successfully performed accurately. Then the results of verification by calculating the keff in the Triga 2000 Reactor code using the converted fissile material data also show the appropriate results. So, this computer code can be used to convert fissile material data from ORIGEN2 to MCNP6 and improve the accuracy of neutronic calculations.

REFERENCES

[1] Spanier J, Gelbard EM. monte carlo principles and neutron transport problems. New York: Dover Publication; 2008.
[2] Los Alamos National Laboratory. Monte Carlo N–Particle Transport Code System Including MCNP6.1, MCNP5-1.60, MCNPX-2.7.0 and Data Libraries. vol. 195. Los Alamos, New Mexico: 2006. https://doi.org/10.1038/1951256a0.
[3] Walker DW. An embarrassingly parallel framework for running EGSnrc / BEAMnrc / DOSXYZnrc , FLUKA , MCNP / MCNPX , GEANT4 and PENELOPE on grid and cluster computers 2015.
[4] Croff AG. ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials. Nuclear Technology 1983. https://doi.org/10.13182/nt83-1.
[5] Zheng M, Tian W, Wei H, Zhang D, Wu Y, Qiu S, et al. Development of a MCNP-ORIGEN burn-up calculation code system and its accuracy assessment. Annals of Nuclear Energy 2014. https://doi.org/10.1016/j.anucene.2013.08.020.
[6] Chang GS. MCWO - Linking MCNP and ORIGEN2 for fuel burnup analysis. Monte Carlo 2005 Topical Meeting, 2005.
[7] Srinath KR. Python – The Fastest Growing Programming Language. International Research Journal of Engineering and Technology (IRJET) 2017;4:354–7.

Preliminary Review for Software Aging on Systems Important to Safety

Catur Febriyanto S.[1, a)] and Farid Noor Jusuf[2, b)]

[1]*Directorate of Nuclear Installations and Materials Regulation – Nuclear Energy Regulatory Agency*
Jl. Gajah Mada no. 8, Jakarta Pusat 10120 DKI Jakarta, Indonesia.

[2]*Center of Nuclear Installations and Materials Regulatory Assessment – Nuclear Energy Regulatory Agency*
Jl. Gajah Mada no. 8, Jakarta Pusat 10120 DKI Jakarta, Indonesia.

[a)] Corresponding author: c.febriyanto@bapeten.go.id
[b)] f.jusuf@bapeten.go.id

Abstract. BAPETEN Chairman Regulation (BCR) No. 8 Year 2008 on Safety Provisions for Non-Power Reactors Aging Management is addressing the aging phenomenon with emphasizing on physical aging. In contrast, non-physical aging, such as software aging, has not been described in detail. The software aging management on the system important to safety ensures that the systems have sufficient safety margins during their life cycle. This review will discuss several recommendations to improve the regulation by comparing several relevant references related to software aging. The results of the review conclude that it is necessary to specify aging indicators used to monitor software aging to predict and forecast software aging before software failure occurs. Besides, it is also necessary to establish the stages of the software life cycles. The purpose of the establishment of the stages is to avoid overlapping definitions of the development and operation of nuclear installations legally, due to differences in characteristics between software aging (non-physical) and physical aging. conclusions also may be used as recommendations, if the BCR will be revised to take into account non-physical aging in more detail in the Appendix of the regulation.

Keywords: software aging management, software life cycles, aging indicator

INTRODUCTION

The continued development of digital technology in nuclear reactors has resulted in new safety and licensing issues, since the existing licensing review criteria were mainly based on the analog devices used when the plants were designed. On the industry side, a consensus approach is needed to help stabilize and standardize the treatment of digital installations and upgrades while ensuring safety and reliability. On the regulatory side, new guidelines and regulatory requirements are needed to assess digital upgrades. Upgrades or new installation issues always involve the potential for system failures [1]. For enrichment facilities, instrumentation and control system is used to monitor enrichment levels based on the ratio of hydrogen to uranium at a certain value. Along with the operation of the facility, of course, the computer code and data will change. These changes should be controlled by applying high standards [2]. If these changes are not anticipated from the beginning, then these changes may jeopardize the safety of the installation by leading to a criticality accident. The criticality accident may lead to contamination and overexposure of workers due to radioactive substances released.

Article 48 Paragraph (4) of Government Regulation (GR) No. 2 Year 2014 stipulates that if the licensee will submit an extension of the operating license by attaching several documents as follows safety analysis report, periodic safety assessment report, operational report, and aging assessment report [3]. If there is a change in one of the documents after performing analysis, the change also applies to the other document. This is because those changes also have an impact on operational limits and conditions; operating procedures; maintenance, surveillance, and inspection programs; as well as aging management programs.

IAEA has described in Requirement 37 that in the design stage, the aging management should be taken into account the life cycles of the technology used and the possible obsolescence of the technology [4]. Furthermore, computer-based equipment in systems important to safety through the development process, including control, testing, and commissioning of design changes, should be taken into account all phases of the life cycles of computer-based systems. Reliability analysis is a major factor in aging management. Currently, RSG - GAS has used its operational reports optimally to analyze the reliability of structures, systems, and components (SSCs) [5]. The operational data in the operational report is used to obtain data on the failure and maintenance of the sub-system. The method used in the analysis is a comparison of the average failure time to the specified maintenance time intervals. Given that the aging management of digital control systems is a relatively new topic, no significant research on this topic has been established [6].

An aging detection plan as required in Article 18 of GR No. 54 Year 2012 is carried out by collecting and analyzing data related to the aging of the SSCs before the commissioning stage began [7]. The aging detection plan is outlined in the aging management program by taking into account the obsolescence of the instrumentation and control systems, as required in Article 11 Paragraph (4) of BCR No. 8 Year 2008 [8]. Nevertheless, the BCR has not stipulated for aging due to technological changes, particularly software. Because of the requirements in the BCR mostly stipulate for physical aging of the SSCs. Similarly, IAEA describes that the obsolescence evaluation for software should be carried out. However, a detailed description of the method for evaluating software obsolescence is not described [9]. Generally, modifications to the instrumentation and control systems may occur during the lifetime of the facility. Considerations for modifications on the system should be justified, for instance in changing from one technology to another (such as, changing from an analog system to a digital system or the obsolescence of the existing instrumentation and control system leading a lack of spare parts).

This paper will discuss software aging management emphasizing systems important to safety, as well as software aging information needed during the software development life cycles. The result of the review is expected to be considered as recommendations in the revision of BCR No. 8 Year 2008 in the future.

METHODOLOGY

The method used in this review is a qualitative method by a literature study using relevant references. The references are then compared with the corresponding requirements in the BCR. The references used in this review are documents issued by the IAEA, other papers, and guidelines published by other countries. By comparing the relevant references, it is expected to obtain a more comprehensive review to strengthen the regulatory framework on requirements for computer-based systems important to safety in adapting to technological development.

RESULT AND DISCUSSION

Software Aging

The advantages of using computer-based systems are automatic control, quick time response, and low operational costs. Digital control systems can implement proactive measures to maintain the safety of nuclear reactors, as well as provide accurate information to the operator.

In general, the service life of electronic and electrical systems and components is less than the lifetime of the facility. Aging mechanisms that can affect the components of instrumentation and control systems, and the measures to understand the effects of these mechanisms should be identified during design. Potential significant aging effects should be addressed to demonstrate that the required safety functions are maintained until the end of service life. Further conservatism should be provided to allow unanticipated aging mechanisms [10]. Aging management for digital control systems of nuclear reactors ensures that the systems are operating in an adequate safety margin throughout their life cycles [11]. Software aging management is crucial because the software is the core difference between digital and analog systems. Consequently, more attention needs to be paid to the software aging management.

Unplanned computer system outages are more likely to be the result of software failures than of hardware failures. By continuous execution of the software for a long period, it will decrease the performance and/or increase the rate of hang/crash failures [12]. Generally, software degradation is characterized by memory bloating and leaking, unreleased file-locks, data corruption, storage space fragmentation, and accumulated round-off errors [13, 14]. Michael Grottke, et al. have classified the aging effects based on common characteristics, as described in **TABLE 1** [15]. Besides, the aging effects may also be classified into volatile and non-volatile effects. They are considered volatile if they are removed by re-initialization of the system or process affected, for example via a system reboot. In contrast, non-volatile aging effects still exist after the reinitializing of the system or process.

Aging software is influenced by several main factors, including [16]:

- Functional factor, related to software usage,
- Environmental factor, related to external factors involving accessories, alternatives, and technological change,
- The human factor, related to the environment, staff, users, education, training, and popularity, and
- Historical factors, related to the acquisition, time of purchase, time of manufacture, technology, and the age of software.

TABLE 1. Classes of aging effects

Basic class	Extension	Examples
Resource leakage	(1) OS – specific (2) App – specific	- Unreleased ● *Memory* (1, 2) ● *File handler* (1) ● *Sockets* (1) - Unterminated ● *Processes* (1) ● *Threads* (1, 2)
Fragmentation	(1) OS – specific (2) App – specific	- Physical memory (1) - File system (1) - Database files (2)
Numerical error accrual	(1) OS – specific (2) App – specific	- Round-off (1, 2)
Data corruption accrual	(1) OS – specific (2) App – specific	- File system (1) - Database files (2)

Software Life Cycles

Generally, digital control systems include hardware and software which are integral and inseparable. By considering the ability of software to handle complex logic and calculation functions, the physical limitations of the hardware can be overcome. In contrast, attention to the complexity of the safety systems should be avoided both in the functionality of the systems and in its implementation by complying with a structured design. By avoiding the complexity in design, it may minimize the use of components, simplify the interface, simplify the verification and validation, and simplify the maintenance of hardware and software.

Instrumentation and control digital systems should be designed by taking into account reliability, aging, safety and security interface, qualification of equipment, easiness for testing, and easiness for maintenance [6, 10]. Besides, the design of the systems also takes into account the concept of defense in depth [10, 17]. The implementation of defense in depth in the system, such as a protection system, will actuate after control systems fail. These considerations have complied with Article 9 and Article 11 of GR No. 54 Year 2012, in which the design of the SSCs should comply with the basic principle requirements of nuclear safety and general design requirements [7].

The other advantages of computer-based systems are the easiness of detecting, localizing, and diagnosing a potential and actual failure, so that the systems may be repaired and replaced efficiently. Otherwise, failures of the digital system software are difficult to intuitively check, classify, and correct, which may affect performance [6]. Because software failures are systematic and not random unlike hardware failures, so software errors and defects cannot be completely discovered through exhaustive testing.

The aging assessment in the BCR No. 8 Year 2008 includes the screening of SSCs, surveillance programs, data collection, and evaluation of aging [8]. Besides, at the screening stage, SSCs are grouped into four groups whose definitions are less suitable when implemented in the software aging management. IAEA also recommends stages for instrumentation and control systems and equipment including design, construction, commissioning, operation, and maintenance. When the stages in the BCR No. 8 Year 2008 and the documents issued by IAEA will be taken into account in the software aging management, of course, it will be difficult to be implemented due to different characteristics with physical aging. Besides, if the stages as recommended by the IAEA will be taken into account, by definition is less appropriate when compared with the definition of the development and operating stages of the nuclear installations in GR No. 54 Year 2012 and GR No. 2 Year 2014.

Therefore, it is necessary to establish the stages of aging assessment particularly for software aging in the BCR. Liang, et al. have studied software aging and propose the software life cycles including design, implementation, testing, operation, and maintenance stages, as described in Figure 1. Due to the stages of software life cycles are different to the stages in the development and operating of nuclear installations in GR No. 54 Year 2012 and GR No. 2 Year 2014, so that overlapping on the definition of development and operating of nuclear installations may be avoided legally.

Similar to hardware aging, the software also needs to be tested and reviewed to reduce software defects, but this regard cannot guarantee that the software system is faultless. If the hardware fails, they may be repaired or replaced. While the associated software should be redesigned or upgraded. Nevertheless, redesigning and upgrading may introduce a new design defect.

Aging on a digital system does not cause significant problems in design because digital equipment is installed in a moderate environment and because it is accessible for monitoring, calibration, and replacement [1]. Although this regard does not cause significant design problems, aging behavior in some software needs to be observed through one or more indicators of aging. The indicator of aging depends on the time since the last rejuvenation. Besides, it also depends on other metrics such as the number of requests processed, the number of database operations performed, the size of the swap file, or the amount of main memory used by the software [18]. The purpose of determining these indicators is to monitor and determine trends in the effects of aging [19, 20, 21].

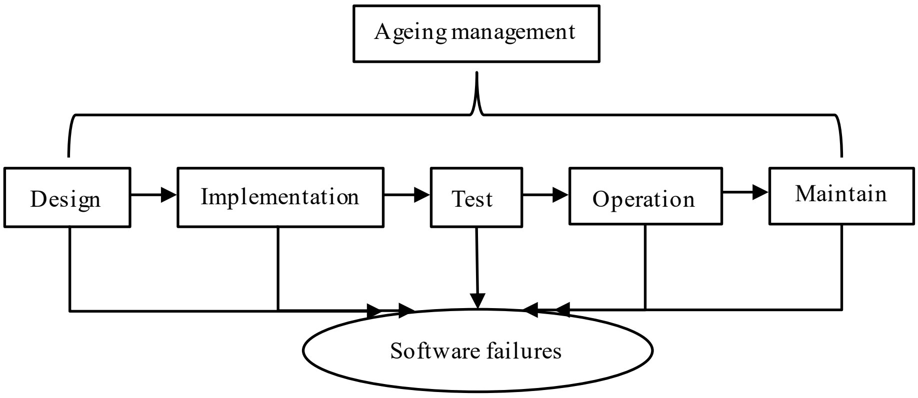


FIGURE 1. Life cycle of software

Indicator for Software Aging Management

When referred to Article 13 Paragraph (2) of BCR No. 8 Year 2008, several specific work factors that can accelerate aging including humidity, fluid chemistry (pH, conductivity), radiation, temperature, pressure, vibration and rotation, and flow. However, these factors are less suitable when applied to software aging. Until now, several researchers have conducted studies in determining the indicators of aging. Liang, et al. has conducted a study related to the information needed in software aging management and the methods for collecting the information for each phase of the software life cycle. Information for software aging management needs to be collected during the design phase through to maintenance phase that will be used to predict and evaluate software aging. Meanwhile, Michael G., et al. classify indicators of aging into two general classes according to their granularity [15]:

1. System-wide indicators provide information related to subsystems shared by several running applications. Indicators in this category are often used to evaluate the effects of aging on the system as a whole and not for special applications, since the shared nature of their environment may cause noise in the captured data. For example, free physical memory, used swap space, file table size, and system load.
2. Application-specific indicators provide information about individual application processes, which provide more accurate information than system-wide indicators. For example, the resident set the size of the process and response time.

Thamarul I., et al. in their literature study summarize that the use of a mitigation approach for aging software is more widely used compared to the aging detection and a combination of both. The mitigation approach involves introducing and suggesting the best suitable solution to delay the phenomenon of software aging. Whereas, the aging detection involves observing and monitoring software systems to allocate the potential for unusual behavior in the software. Mitigation in the aging effect here means operations, maintenance, repair, and replacement actions to mitigate detected aging effects and/or degradation of the structure or component [22]. Of course, the detection and mitigation of the aging effects should be carried out to comply with the requirement as recommended by the IAEA [4]. In other words, the detection and prevention of aging effects applied at nuclear installations should be carried out in sequence.

Besides, the literature study is also summarized that several types of analysis for aging detection, such as measurement-based is more widely used than model-based and hybrid. The measurement-based analysis involves the statistical approach by monitoring the software/system behavior directly using aging indicators to predict and forecast the aging phenomenon. Meanwhile, model-based analysis involves adopting stochastic prosses to model the aging phenomenon. Those indicators which have been determined are used as inputs in analysis for software aging detection. One of the results of the analysis is scheduling software rejuvenation which is a part of the mitigation approach (prevention of software aging) [18, 23].

CONCLUSIONS

In this review, it can be concluded that the description of aging management should be in detail, not only for physical aging but also for non-physical aging, particularly for software aging. Even though, non-physical aging is described in the Main Text of the regulation, which is the implementation of an aging management program taking into account the obsolescence of instrumentation and control system, and other conditions such as technology changes. Therefore, it should be described in detail in the Appendix of the regulation related to indicators used for monitoring software aging to predict and forecast software aging. Besides, it should be established the stages of the life cycle of software, due to the difference of the characteristics of software aging (non-physical) compared to physical aging. Moreover, the establishment of the stages is to avoid overlapping on the definition of development and legally operating nuclear installations. Those conclusions may be used as recommendations, if BCR No. 8 Year 2008 on Safety Provisions for Non-Power Reactors Aging Management will be revised in the future.

REFERENCES

[1] NEA – OECD, Operating and Maintenance Experience with Computer-Based Systems in Nuclear Power Plants, A report by the PWG-1 Task Group on Computer-based Systems Important to Safety, France, 1998

[2] IAEA, Safety of Nuclear Fuel Cycle Facilities, Specific Safety Requirements (SSR-4), Vienna, 2017

[3] Government Regulation, Licensing for Nuclear Installations and Nuclear Material Utilization, Government Regulation No. 2 Year 2014, Jakarta, January 2014

[4] IAEA, Safety of Research Reactors, Specific Safety Requirements (SSR-3), 2016, Vienna

[5] Mike S., Enrin H., Aep A., Purwadi, Analisis Keandalan Perawatan Sub-sistem Elektrik, Instrumentasi dan Kontrol RSG – GAS, Sigma Epsilon Vol. 23 No. 2, November 2019, pp. 79 – 90, Jakarta (in Bahasa)

[6] Liang, et al., Discussion on Software Aging Management of Nuclear Power Plant Safety Digital Control System, SpringerPlus (2016) 5:2092

[7] Government Regulation, Safety and Security for Nuclear Installations, Government Regulation No. 54 Year 2012, Jakarta, April 2012

[8] BAPETEN Chairman Regulation, Safety Provisions on Aging Management for Non-Power Reactor, BAPETEN Chairman Regulation No. 8 Year 2008, Jakarta, March 2008

[9] IAEA, Ageing Management for Research Reactors, Specific Safety Guide (SSG-10), 2010, Vienna

[10] IAEA, Instrumentation and Control Systems and Software Important to Safety for Research Reactors, Specific Safety Guide (SSG-37), 2015, Vienna

[11] Simola K., Reliability Methods in Nuclear Power Plant Ageing Management, Dissertation, 1999, Espoo

[12] Thein T., Chi S. D., Park J. S., Proactive Fault Management with Virtualization for Software Aging, Software Engineering, Artificial Intelligence, Networking and Parallel/Distributed Computing, Springer, pp. 408 – 413, Berlin, 2008

[13] Garg S., Van Moorsel A., A Methodology for Detection and Estimation of Software Aging, In: The Ninth International Symposium on Software Reliability Engineering, IEEE, pp. 283 – 292, Washington, DC., 1998

[14] Castelli V., et al., Proactive Management of Software Aging, In: IBM Journal of Research and Development Vol. 45, No. 2, New Jersey, 2001

[15] Michael G., et al., The Fundamental of Software Aging, In Proc. 1st International Workshop on Software Aging and Rejuvenation, 19th International Symposium on Software Reliability Engineering, 2008

[16] Jamaiah Y., et al., Software Ageing Measurement and Classification Using Goal Question Metric (GQM) Approach, Science and Information Conference 2013, pp. 160 – 165, London, 2013

[17] IAEA, Configuration Management in Nuclear Power Plants, IAEA-TECDOC-1335, Vienna, 2003

[18] Artur. A. and Luis S., Deterministic Models of Software Aging and Optimal Rejuvenation Schedules, CoreGRID Technical Report Number TR-0047, Institute on System Architecture, 2006

[19] NEA – OECD, Ageing Management of Nuclear Fuel Cycle Facilities, CSNI Technical Opinion Papers No. 15, French, 2012

[20] IAEA, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, Specific Safety Guide (SSG-48), 2018, Vienna

[21] Thamaratul I., et al., Risk Mitigation for Anti Software Ageing – A Systematic Literature Review, Journal of Theoretical and Applied Information Technology, Volume 97 Number 14, 2019

[22] HSE, Management of Ageing – A Framework for Nuclear Chemical Facilities, RR912 Research Report, 2012

[23] Kalyanaraman V. and Kishor S. Trivedi, A Comprehensive Model for Software Rejuvenation, IEEE Transactions on Dependable and Secure Computing, Vol. 2 No. 2, 2005

Fukushima Plus: Testing for Public Safety

Lars Jorgensen[1, a)] and Robert Hargraves[1, b)]

[1] *ThorCon US 142 Lyons Road, Stevenson, Washington, USA 98648*

[a)]Corresponding author: ljorgensen@thorconpower.com
[b)]rhargraves@thorconpower.com

Abstract. Next generation nuclear power plants are required to exceed the safety of existing nuclear power - which is already the safest form of electricity generation. The dominant harm from nuclear power plant accidents is not the radiation itself but rather the poor public response driven primarily by fear. ThorCon intends to address this fear by demonstrating on a live reactor an immediate full station blackout at full power with no operator intervention. This is a more severe external event than occurred at Fukushima Daiichi which had full power cooling for 45 minutes after reactor shutdown before the station blackout. The goal of the demonstration is to build confidence in the regulator, government, and press that an evacuation is not necessary even in very severe external events. Such a severe test naturally raises concerns about the safety of the test itself. This paper presents a series of smaller tests, each building on the previous tests leading up to the final live test to ensure that the test itself is not unsafe. Key Words: panic evacuation, safety case, plant description, four loops, primary loop, drain tank, coldwall, Fukushima Plus.

FEAR: THE MOST DANGEROUS ELEMENT OF NUCLEAR POWER

In terms of lives lost nuclear power is already the safest form of electricity generation.(1) But for next generation nuclear we are called on to be even safer than current nuclear. To see where there is room for improvement we should examine the cause of previous accidents and the harm that resulted. We have three major accidents to study. Common to all is that it took several errors to lead to the accident.(2) Also common is that human errors either in design or during the accident were major contributors.(3) In each accident the number of people hurt or killed by radiation was a small fraction of the harm that resulted from the accident.

The most recent event occurred at the Fukushima Daiichi plant in Okuma. A few workers received enough radiation to warrant some treatment and on-going health monitoring.(4) One person died of cancer that may be related to radiation from the plant. Japan took the proper precautions regarding iodine to protect the public. The radiation levels received by the public were so low that any health effects are expected to be too small to detect.(5) However, this is not to say there wasn't substantial harm. Around 1,600 people died due to the panic evacuation.(6) The poor response to the accident caused much more harm than the accident itself. The same was true of the even worse accident at Chernobyl.

As we expand nuclear power we must reduce the ability of human error to cause an accident. This is generally acknowledged in the industry as we move to passive safety systems. However, even more important is to reduce the panic response in an accident. The vast majority of the harm from nuclear power is from panic responses not accidents themselves. We must work to build public and government confidence in the safety of nuclear power. Building confidence in the safety of nuclear is an activity that requires action on the part of the developer, operator, regulator, legislator, and president.

The safety case for ThorCon is built primarily on live testing. Hearing that we are going to deliberately cause events like happened at Fukushima definitely raises some concerns. This paper will provide an overview of the tests planned leading up to the Fukushima Plus test to ensure that conducting such a test is safe. ThorCon plans to conduct the final test in the presence of the press and government to reassure them that even with an external event worse than Fukushima an evacuation is not appropriate.

Plant Description

ThorCon is a molten salt fission reactor. Unlike all current nuclear reactors, the fuel is in liquid form, which can be moved around with a pump and passively drained. ThorCon is a straightforward scale-up of the successful Molten Salt Reactor Experiment (MSRE) at the Oak Ridge National Laboratory, United States. A full-scale 500 MWe ThorCon prototype can be tested within four years, that is, by 2025. After proving the plant safely handles multiple potential failures and hazards, commercial production can begin.

A ThorCon plant requires less of the planet's resources than a coal plant.(7) Assuming efficient, evidence-based regulation, ThorCon can produce clean, reliable, CO2-free electricity at US$0.03/kWh — cheaper than coal. The complete ThorCon plant is manufactured in 150 to 500 ton blocks in a shipyard. This produces order of magnitude improvements in productivity, quality control, and build time. A single large reactor yard can turn out twenty gigawatts of ThorCon power plants per year.

A ThorCon power plant is built in a shipyard, then towed to a nearshore site and ballasted down onto the seabed. The sea provides for transport of the complete power plant and the provisioning of fuel and reactor Cans. It also provides water for steam condenser cooling. **FIGURE 1** below shows a CanShip and two 500 MWe units with one undergoing a maintenance cycle that occur once every four years.

ThorCon is a high temperature reactor with thermal efficiency of 46.4% compared to about 33% for a standard light water reactor, reducing capital costs and cutting cooling water needs by 40%. ThorCon employs four flow

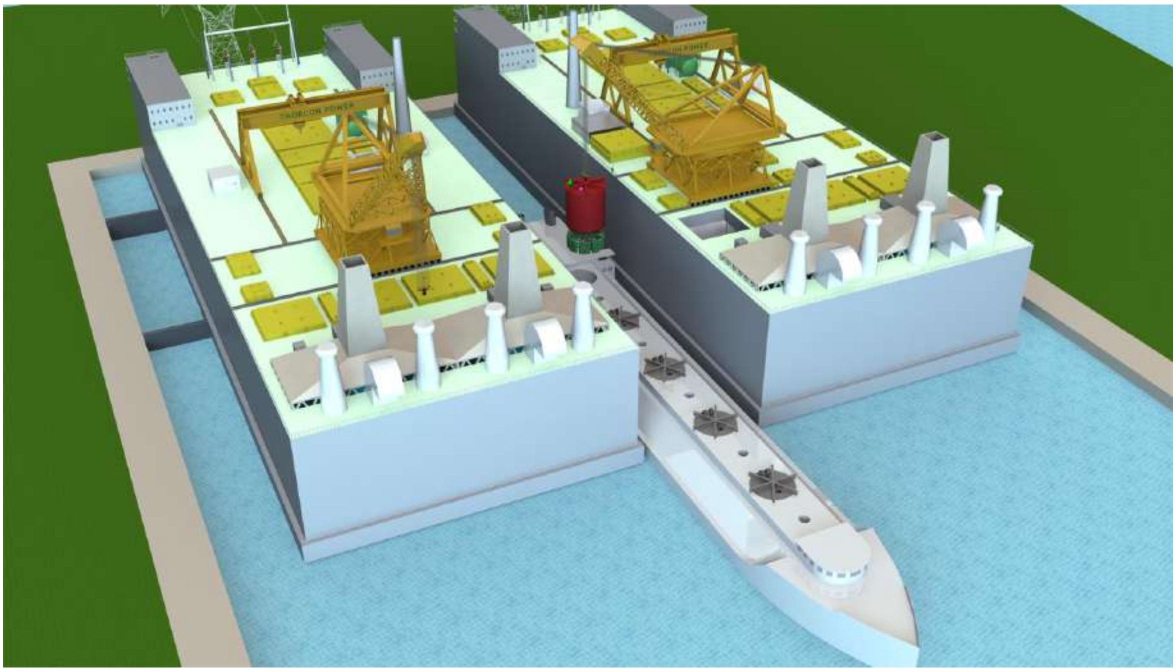


FIGURE 1. 2 x 500 MWe ThorCon power units with Can and fuelsalt service ship

loops for converting nuclear heat to electric power, 1) the primary loop inside the Can, 2) the secondary salt loop, 3) a solar salt loop, and 4) a supercritical steam loop. The ThorCon steam loop is a standard, single reheat, super-critical steam cycle, nearly off the shelf technology.

ThorCon does not rely on electric power from the grid for startup or any electricity for safety. The plant can load-follow, handle disconnects, self-start, and also help blackstart a powerless grid. Regular maintenance will occur at 4-year intervals, when a CanShip visits to exchange fuel casks and Cans.

The primary loop is shown in **FIGURE 2**. Fuelsalt flows through the primary loop containing the reactor Pot, the primary loop pump (PLP), and the primary heat exchanger (PHX). The graphite moderator in the Pot contains channels through which fuelsalt flows up. Fuelsalt enters the Pot at 565°C and leaves at 704°C. Pot pressure is 3 bar gage, about the same as a garden hose. The thermal energy output per Can is 557 MWth. The Can is a cylinder 10.3 m high and 7.8 m in diameter, weighing about 343 tons. The Can has only one major moving part, the primary loop pump.

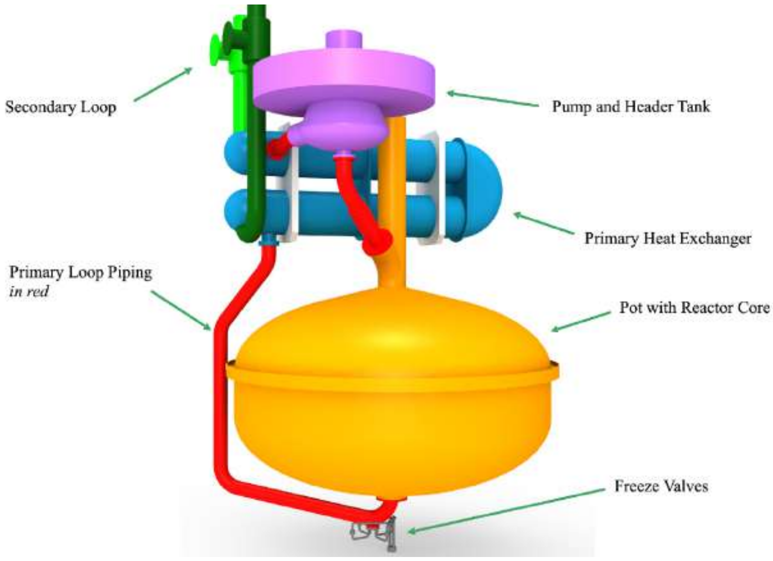


FIGURE 2. Primary Loop with Pump, Heat Exchanger and Pot

Below the Pot is a freeze valve. The freeze valve stays closed using active cooling. It will open within ten minutes by melting should the cooling ever stop (such as happens with station blackout).

FIGURE 3 below shows the primary loop surrounded by the Can (in red), and below it the Fuelsalt Drain Tank (FDT) in green. The drain tank is composed of 32 3.5m tall, 0.5m diameter cylinders which spreads the fuelsalt out far enough that criticality is not possible and provides a large surface area to volume ratio which facilitates extraction of decay heat. The FDT is designed to handle the substantial contraction of salt freezing and expansion on thawing.

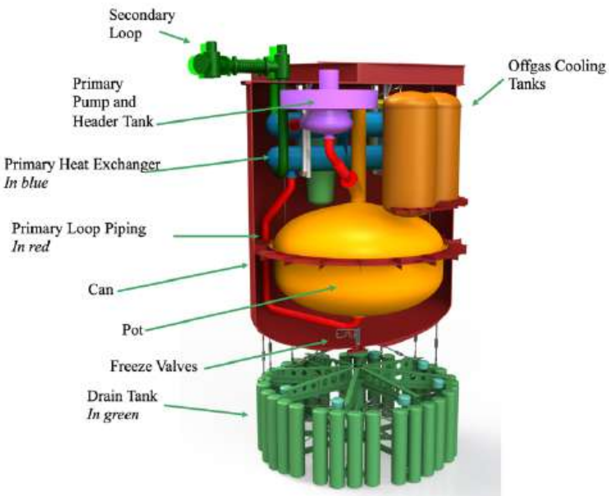


FIGURE 3. Fuelsalt Drain Tank and Can with Primary Loop Inside

Decay heat radiates from the FDT to the Coldwall, shown in dark blue in Figure 4. The heat transfer is proportional to the absolute temperature to the fourth power. The Coldwall consists of 25mm steel, 500mm water, and 25mm steel. As the radiated heat brings the steel up in temperature the water inside the Coldwall boils and the water/steam mixture rises rapidly creating strong natural circulation. The steam/water mixture goes up through a riser to the condenser (far right) where the heat is released and the steam converted back to water. The cooled water is then returned to the basement and eventually re-enters the Coldwall at the bottom in a closed loop system. The condenser is located in the bottom of the Cooling Pond. The cooling pond water is evaporated to disperse the decay heat into the atmosphere. The cooling pond contains sufficient water to dissipate the decay heat for 145 days before it drys out.

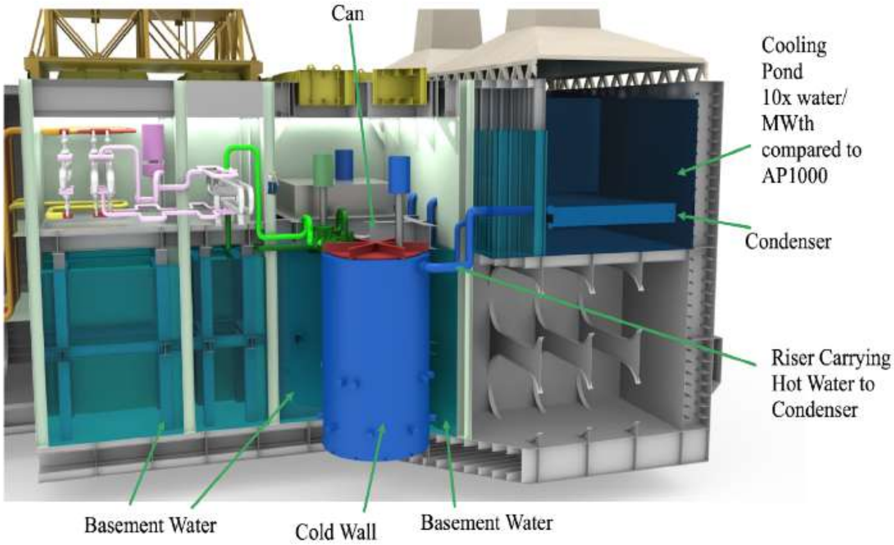


FIGURE 4. Coldwall, Condenser and Cooling Pond Removes Decay Heat for 145 Days

TEST PLAN FOR FUKUSHIMA Plus

(Immediate Station Blackout)

Step 1: Pre-fission test platform, normal temperature, no decay heat.

Testing starts with a pre-fission test platform. This test platform contains a full scale Can/FDT and Coldwall. Electrical heaters external to the Pot (reactor vessel) are used to heat up the fuel salt to normal operating temperature (635°C). The fuelsalt in the pre-fission test platform contains no fissile - all heat is supplied by electrical heaters. A drain is performed and the temperature distribution on the walls of the FDT are recorded versus time. Heat removed via the Coldwall is also recorded. The results are compared with simulation results to validate the software simulations.

Step 2: Pre-fission test platform, high temperature, simulated decay heat.

Next the fuelsalt is heated to the maximum temperature achievable in the Pot then a drain is initiated. In the FDT heaters are used to simulate decay heat. Again, the temperature distribution and heat extraction are recorded versus time and compared with software predictions. If the temperatures deviate significantly from predictions the test is easily aborted by turning off the electrical heaters which eliminates risks in this test. Decay heat is well understood so a close upper bound on decay heat can be reliably predicted. The test will last until the FDT is cool enough that creep damage is essentially over. We anticipate this will be within four hours. Once the test is successfully completed we will have shown that the FDT and decay heat removal system can handle a drain of fuel salt from full power.

Step 3: Repeat Step 2 Using the Demonstration Plant

Once tests are completed at the pre-fission test platform a 500 MWe demonstration plant will be built and installed. As part of the initial testing before any fission has occurred step 2 will be repeated to ensure that the demonstration plant performs just like the pre-fission test platform.

Step 4: Fission Testing Using the Demonstration Plant

After producing partial power (for example 10%) for a short time (for example 15 minutes) repeat the test. The decay heat produced initially is proportional to the power level. Operating for a short time ensures that the decay heat will reduce much more rapidly than after fission products have been allowed to build up for a long time. Repeat the test in steps increasing the power level and fission product inventory, verifying at each step that the measured data reasonably matches the predictions. Electrical heaters in the FDT can be used to augment the decay heat to mimic the decay heat produced by a full fission product inventory after running the power plant at full power for four years. Again, since the additional heat is supplied by electricity if the test deviates significantly from expectations the electrical heat can be removed. The Test Approval Committee must authorize each group of steps.

Step 5: FukushimaPlus Testing Using the Demonstration Plant

Once confidence has been built through the previous tests the reactor can be run at full power for a significant time and a test executed. The press, and government officials will be invited to witness repeat testing so that they can be confident that even with an immediate, full station blackout there is no need for an evacuation.

CONCLUSION

As part of the effort to reduce unreasonable fear of nuclear power we intend to conduct a series of tests culminating in a demonstration on a live power plant an immediate full power, full station blackout with full passive safety (no operator, electricity, or machinery action). This is a much more severe test than the events that occurred at Fukushima. This will be done to provide the regulator, press, public, and government confidence that an evacuation is unnecessary in the event of a future severe external event such as an earthquake or tsunami.

REFERENCES

[1] Hannah Ritchie (2014) - "Energy". *Published online at OurWorldInData.org.* Retrieved from: 'https://ourworldindata.org/energy' [Online Resource]

[2] World Nuclear Association Study. Retrieved from: https://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/three-mile-island-accident.aspx https://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/chernobyl-accident.aspx https://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/safety-of-nuclear-power-reactors.aspx

[3] World Nuclear Association Study. Retrieved from: https://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/three-mile-island-accident.aspx https://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/chernobyl-accident.aspx https://www.world-nuclear.org/information-library/safety-and-security/safety-of-plants/safety-of-nuclear-power-reactors.aspx

[4] UNSCEAR 2013 Report to the General Assembly with Scientific Annexes. VOLUME I. Retrieved from: http://www.unscear.org/docs/reports/2013/13-85418_Report_2013_GA_Report.pdf

[5] UNSCEAR 2013 Report to the General Assembly with Scientific Annexes. VOLUME I. Retrieved from: http://www.unscear.org/docs/reports/2013/13-85418_Report_2013_GA_Report.pdf

[6] Wade Allison, *When Fear Kills: The Case of Nuclear Energy,* 2018. Retrieved from researchgate.net/publication/328996079_When_Fear_Kills_The_Case_of_Nuclear_Energy_pdf_version

[7] ThorCon Team, *ThorConIsle*, 2020. Retrieved from: http://thorconpower.com/docs/docs_v140_isle.pdf

Analysis of The Radionuclides I-131, I-133, Kr-85, And Xe-133 Dispersioon from SMR 100 MW in Beach Bengkayang West.

Wiku Lulus Widodo[1)]

[1] *Pusat Kajian Sistem Energi Nuklir – Batan*

wikululu@batan.go.id

Abstract. Batan has a task to carry out PRN (National Research Programme) activities for strategic planning 2020-2024. This study aims to support the feasibility study of NPP (Nuclear Power Plant) in West Kalimantan which is part of Batan activity for PRN. The potential for the spread of radionuclides from an NPP is an indicator of whether an NPP is feasible to build in Kalimantan. The objective of this study is to calculate and analyze the distribution of radionuclides from NPP under normal operating conditions with locations in West Kalimantan which are expected to be built in Gosong beach the Bengkayang region. In this study, MELCOR and SIMPACT software is used to simulate radionuclide emissions from NPP and their distribution to the area around the NPP. Radionuclide distribution analysis has been carried out from a 100 MW nuclear power plant facility. Radionuclides analyzed were I-131, I-133, Kr-85, and Xe-133 which are fission products from nuclear reactions. Radionuclide distribution calculations using SIMPACT software released by the IAEA. The calculation results show that the dispersion of I-131 has a maximum exposure of 0.0179 Bq/m3 with a range of 50 km to the east, 80 km to the north, and 45 km to the south. The dispersion of I-133 has a maximum exposure of 0.0280 Bq/m3 with a range of 45 km to the east, 70 km to the north, and 35 km to the south. The dispersion of Kr-85 has a maximum exposure of 1.72 x 10-3 Bq/m3 with a range of 85 km to the east, more than 100 km both north and south. The dispersion of Xe-133 has a maximum exposure of 3.15 x 10-2 Bq/m3 with a range of 90 km to the east, more than 100 km to the north, and 95 km to the south.

Keywords: Radionuclide Dispersion, SMR, West Kalimantan, I-131, I-133, Kr-85, Xe-13

INTRODUCTION

This study aims to support the feasibility study of NPP in West Kalimantan which is part of Batan's task to carry out PRN (National Research Programme) activities. The potential for the spread of radionuclides from an NPP is an indicator of whether an NPP is feasible to build in Kalimantan. The objective of this study is to calculate and analyze the distribution of radionuclides from NPP under normal operating conditions with locations in West Kalimantan which are expected to be built in Gosong beach Bengkayang. In this study, MELCOR and SIMPACT software is used to simulate radionuclide emissions from NPP and their distribution to the area around the NPP. The issue of radionuclide emissions which has the potential to expose radiation is an important matter related to the community's acceptance of the NPP development program. Open information regarding the potential and level of concentration of radionuclide distribution is highly expected by the general public so that they can assess whether the nuclear power plant development program will be supported or not.

The release of radionuclide emissions into the atmosphere is similar to carbon emissions. There is a challenge today in assessing radiological dose from the nuclear reactor using a more reliable computer tool in addressing the released radionuclide to the atmosphere and ground effectively[1]. For PWR reactors (VVER) radionuclides contribution to the annual dose from atmospheric releases is more complicated, but, in general, the dose is formed by tritium, ^{14}C, and noble gases[2]. An overview of recent studies in which the transport of radionuclides in porous materials has been recently modeled is provided[3]. Radioactive dose dispersion map using the fallout stack model, and Annual effective dose estimation of the mentioned reactor have been studied in wide term[4].

The Fukushima Daiichi Nuclear Power Plant (FDNPP) accident, which occurred in March 2011, has released large amounts of radionuclides into the atmosphere, resulting in the contamination of terrestrial and marine environments[5]. the improved Weather Research and Forecasting Chemistry (WRF-Chem) model is used to study the atmospheric migration process of radionuclides released by HYNPP in a hypothetical accident to assess the potential risks to the public[6]. In the normal operation conditions, the atmospheric dispersion of radioactive material is calculated using CAP88-PC code to estimate the impact of dispersion. A Gaussian dispersion air transport plume model was used to simulate the atmospheric dispersion of radionuclides in different atmospheric stability classes and various wind speeds and directions[7].

Installing new nuclear facilities requires a serious consideration of safety measures throughout all stages. This work evaluates the radiological impact of the expected operations of potential Baiji nuclear power plant (NPP) in accord with the safety requirement's achievement[8]. An atmospheric dispersion model based on the Fields of Regards (FOR) technique was used to reduce the uncertainties of the trajectory model and to improve the accuracy of detective technology. The simulated results generated by the trajectory and atmospheric dispersion models together were agreed better with the measurements compared to those obtained from the trajectory model alone[9]. This method reformulates the time-consuming 3D integral in the dose rate model as a convolution and uses a fast Fourier transform to accelerate its solution. The convolution form provides a new receptor-oriented insight into dose rate estimation that can flexibly describe the radiological response of biological tissues. The proposed method makes no approximations or assumptions, so it is accurate and applicable to arbitrary atmospheric dispersion models and radionuclide distributions[10].

The capacity of the nuclear power plant that will be discussed in this study is 100 MW so that it is included in the SMR (Small Medium Reactor) NPP category. The assumed SMR nuclear power plant in this study is similar to SMART which was designed by the South Korean KHNP[11]. The calculation of the potential for fission products released from the SMART is calculated based on the technical specifications of the reactor fuel device system using MELCOR[12,13].

West Kalimantan is one of the provinces in Indonesia with a capital city in Pontianak located around the Kapuas River. The most recent estimate of the population of West Kalimantan for December 2019 is 5,011,660 (source: kalbar.bps.go.id). Land use in West Kalimantan is dominated by forests, bushlands, and plantations. Residential land in West Kalimantan is still relatively small when compared to the total area of the entire province.

This study consist of several things including a discussion of the SIMPACT software released by the IAEA to help simulate radionuclide emissions from the SMR nuclear power plant. Conclusions and recommendations lead to information on the extent of the distribution of radionuclide emissions and their concentrations so that it becomes information that the PLTN SMR during normal operation will be eligible to be built in West Kalimantan.

THEORY

Nuclear fission products are atomic fragments that remain after atomic nuclei with large atomic numbers undergo nuclear fission. Uranium through nuclear fission will split into two smaller nuclei with several neutrons and be accompanied by the release of heat energy (kinetic energy from the nucleus), and gamma rays[14,15]. These two nuclei with smaller atomic numbers are called fission products. Fission products themselves are usually unstable and are radioactive; because of the relative abundance of neutrons for their atomic number, many of these fission products quickly undergo beta decay. This releases additional energy in the form of beta particles, antineutrinos, and gamma rays. Thus, fission events usually produce beta and gamma radiation, although this radiation is not produced directly by the fission event itself. Figure 1 shows the distribution of fission products for various types of nuclear fuel.

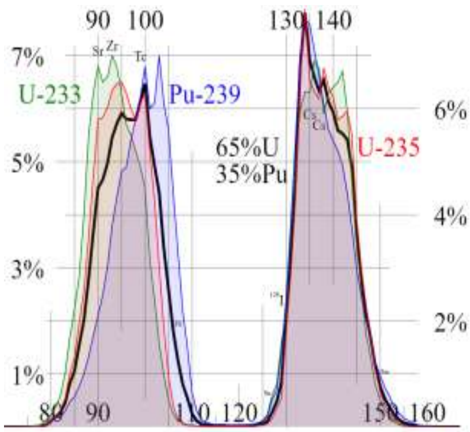


FIGURE 1. The distribution of fission products for various types of nuclear fuel

The analysis was first performed by calculating the reactor inventory. Further calculations were to determine reactor source term for normal conditions using MELCOR with nuclear reactor similar to SMART KHNP. The reactor source term is shown in **TABLE 1**.

TABLE 1. Reactor Source term (Bq)

No.	Radionuclides	Source term (Bq)
1	KR-85	4.72E+9
2	KR-85M	9.99E+10
3	KR-88	2.47E+11
4	I-131	1.18E+12
5	I-132	1.27E+12
6	I-133	2.36E+12
7	I-134	1.18E+12
8	I-135	2.04E+12
9	XE-133	8.55E+11
10	XE-135	3.61E+11

Gas fission products in the temperature and pressure of nuclear reactors will potentially be released as radionuclide emissions. Some radionuclides in the form of gases include I-131; I-133; Kr-85 and Xe-133 so that this radionuclide has the potential to spread to the environment. **TABLE 2** shows the fission properties of Iodine, Krypton, and Xenon. Based on these physical properties, we can see that the radionuclides are gases in the temperature and pressure of a nuclear reactor.

TABLE 2. Types of radionuclides in the form of gases and their physical properties

	Iodine	Kripton	Xenon
Melting Point (^{0}C)	113,7	-157,37	-111,75
Boiling Point (^{0}C)	184,3	-153,42	-108,1
Mass Density (gr/cm^3)	4,933	2,413	2,942
Emitted Radiation	Beta, Gamma	Beta	Beta

(source : https://wikipedia.com)

The Simplified Approach for Estimating Impacts of Electricity Generation (SIMPACTS) model, software developed by the International Atomic Energy Agency (IAEA), adapted the EcoSense method to a simpler form. In the SIMPACTS model, there is one type of module that can be used to estimate the spread of carbon emissions from fossil fuel plants. The method used in this model is the Impact Pathways Analysis (IPA) approach[16][17]. The stages of the IPA method are illustrated in **FIGURE 2**.

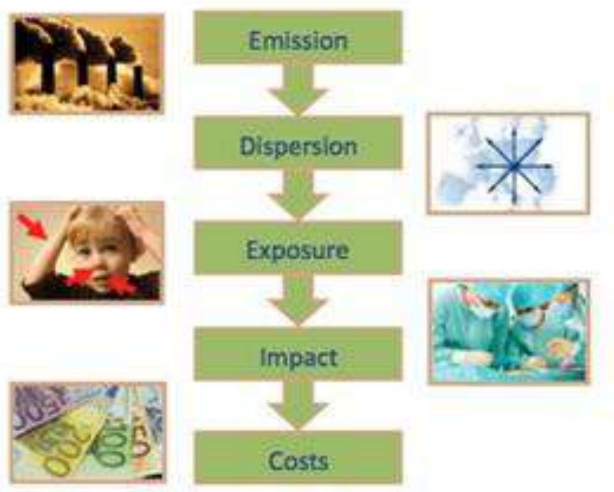


FIGURE 2. Stages of the Impact Path Analysis Method

The initial stage of the IPA method is the identification of the characteristics of the emissions produced by a pollutant source. Some of these characteristics include the location where the source of the emission was produced, the technology used by the emission source, and the type and amount of emissions produced. The second step is the identification of environmental characteristics, dispersion mechanism, and pollutant transportation. At this stage, the calculation of dispersion and transport of pollutants is carried out to produce the value of pollutant concentration (dose), taking into account the weather conditions in the analyzed area. The basic equation used in the calculation of radionuclide distribution is as follows:

$$C = \frac{Q}{2\pi\sigma_x\sigma_y} g \; exp\left[\frac{-d_a^2}{(2\sigma_x^2)}\right] exp\frac{-d_c^2}{2\sigma_y^2}$$

$$g = \frac{2}{(2\pi)^{1/2}\sigma_z}\sum_{n=-\infty}^{\infty} exp\left[\frac{-(H_e+2nh)^2}{(2\sigma_z^2)}\right] C = \frac{Q}{2\pi\sigma_x\sigma_y} g \; exp\left[\frac{-d_a^2}{(2\sigma_x^2)}\right] exp\left[\frac{-d_c^2}{2\sigma_y^2}\right]$$

$$g = \frac{2}{(2\pi)^{1/2}\sigma_z}\sum_{n=-\infty}^{\infty} exp\left[\frac{-(H_e+2nh)^2}{(2\sigma_z^2)}\right]$$

Where,

C : basic concentration (g / m3),
Q : pollutant mass in puff (g),
σ_x : standard deviation of the Gaussian distribution in the direction of the wind (m),
σ_y : standard deviation of the Gaussian distribution perpendicular to the wind direction (m),
σ_z : standard deviation of the vertical direction Gaussian distribution (m),
d_a : distance from the center of the puff to the receptor in the direction of the wind (m),
d_c : the distance from the center of the puff to the receptors that are perpendicular to the direction of the wind (m),
g : the vertical limit of the Gaussian equation (m),
H : effective height above the ground from the center of the puff, and
h : mixing height (m).

The domain area of discussion in this study is West Kalimantan province with its center point being the Gosong beach in Bengkayang Regency. The area analyzed is 200 km x 200 km as shown in **FIGURE 3**.

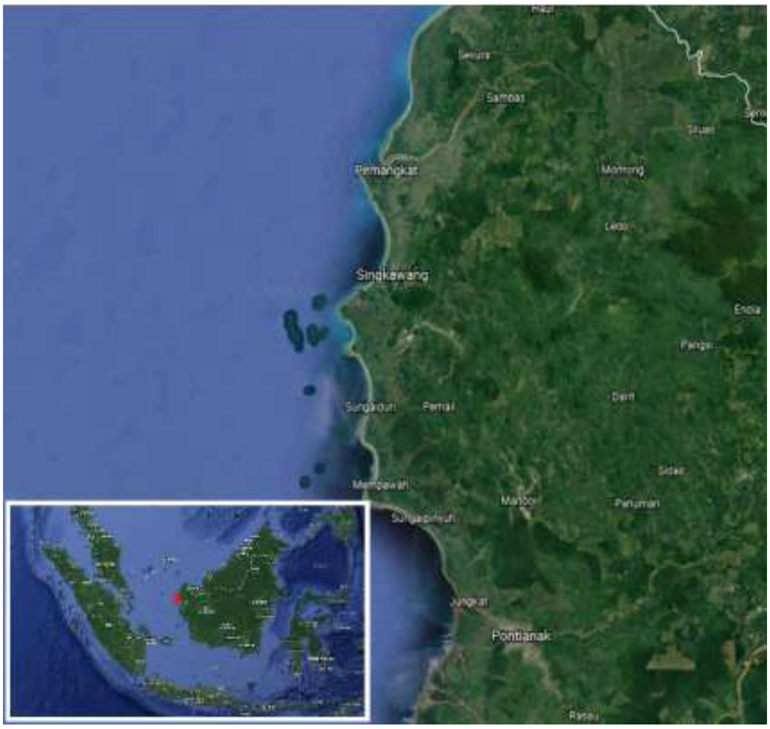


FIGURE 3. Domain area of study about 200 km x 200 km

METHODOLOGY

Calculation of the distribution of carbon emissions which in this study was carried out using SIMPACTS software which is software that was officially released by the IAEA specifically the nuclear energy department (department of nuclear energy) and can be used by member countries for research and development purposes[1819]. There are several stages in the calculation with SIMPACTS, including making cases, inputting data input, including data domains, emission & dispersion, and pathway analysis, and running programs.

The first stage is making cases and domains. At this stage, it was determined that the source of emissions (NPP) was on the west coast of the Kalimantan Island. The area of impact studied is a local scale with an area of 5 km x 5 km in the cell as shown in **FIGURE 4**. This size of the cell is the default from SIMPACT software for local scale options.

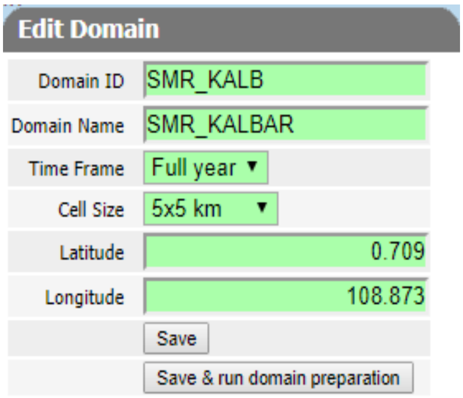


Figure 4. Domain data input

The advantage of the latest SIMPACTS software is that it has integrated topographic data sourced from the GTOPO 30 Project. GTOPO 30 is a digital elevation model derived from several vector topographic information sources[20]. This model has been developed since 1993 at the U.S. Geological Survey's Center for Earth Resources Observation and Science (EROS). In addition, several world organizations also participated in contributing data sources, some of which were the National Aeronautics and Space Administration (NASA), the United Nations Environment Program / Global Resource Information Database (UNEP / GRID), and U.S.Agency for International Development (USAID). The domain display area is shown in **FIGURE 5**.

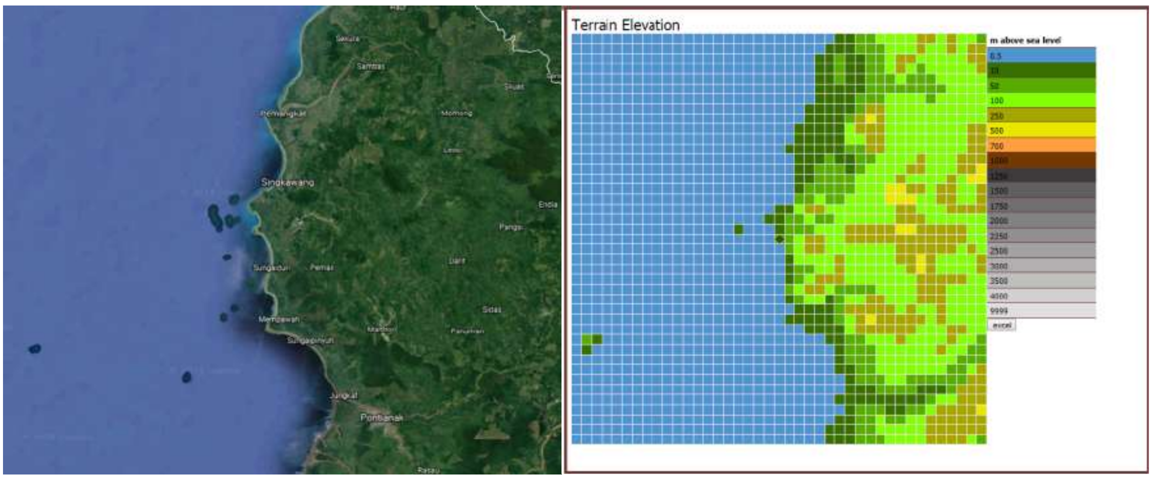


FIGURE 5. Display domain area

The map of the wind direction in the domain of the study area is shown in **FIGURE 6**. The direction of the wind around the location of the nuclear power plant (the middle of the domain area) is dominated to the northwest. The movement of the wind changes its orientation when it passes through an area that has a height above 1000 m above the sea. The direction of the wind in the mid-ocean area is dominated towards the west. All wind directions in the area domain will affect the spread of radionuclide emissions.

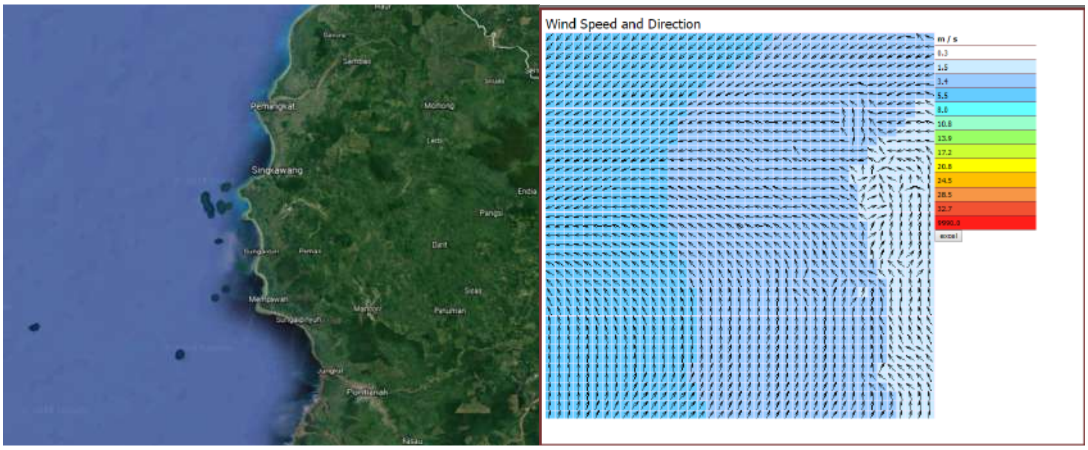


FIGURE 6. Wind direction in the area of the study

The temperature distribution in the domain of the study area is shown in **FIGURE 7**. The temperature around the nuclear power plant has an upward orientation from east to west. Changes in temperature at these locations due to changes in altitude from mountainous conditions leading to open sea waters in the west of the island of Borneo. This temperature distribution also affects the spread of carbon emissions because in general, the temperature will affect the air pressure at a location. The difference in air pressure will affect the direction of the wind which in turn affects the spread of radionuclide emissions.

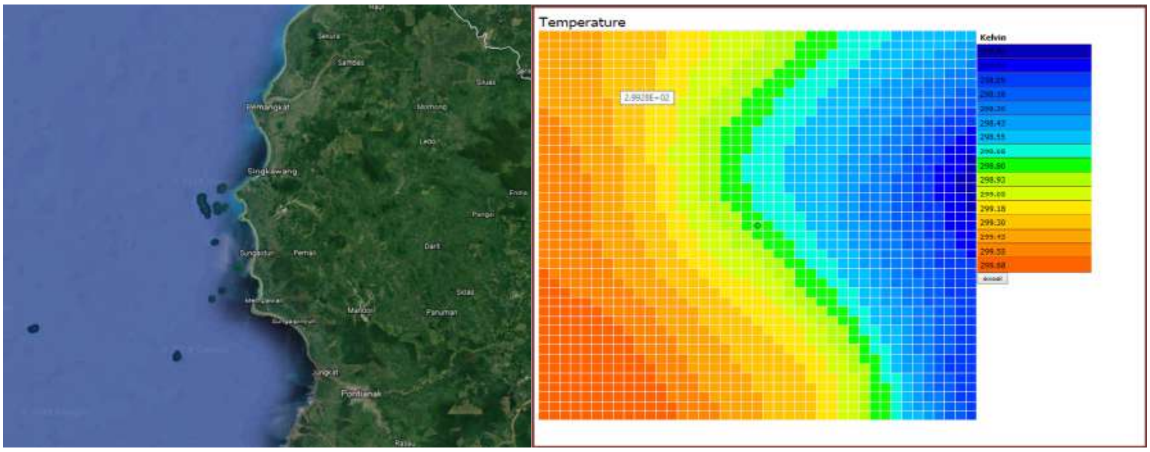


FIGURE 7. Temperature Distribution in the area of the study

The land-use map is shown in **FIGURE 8**. The area around the nuclear power plant is dominated by agriculture and plantations including settlements. Most of the land in the eastern domain area is still forested. Housing and buildings are quite a lot for Pontianak city area, while other areas are still relatively low. Part of the beach is in the form of a wet forest which is possible in the form of Mangrove.

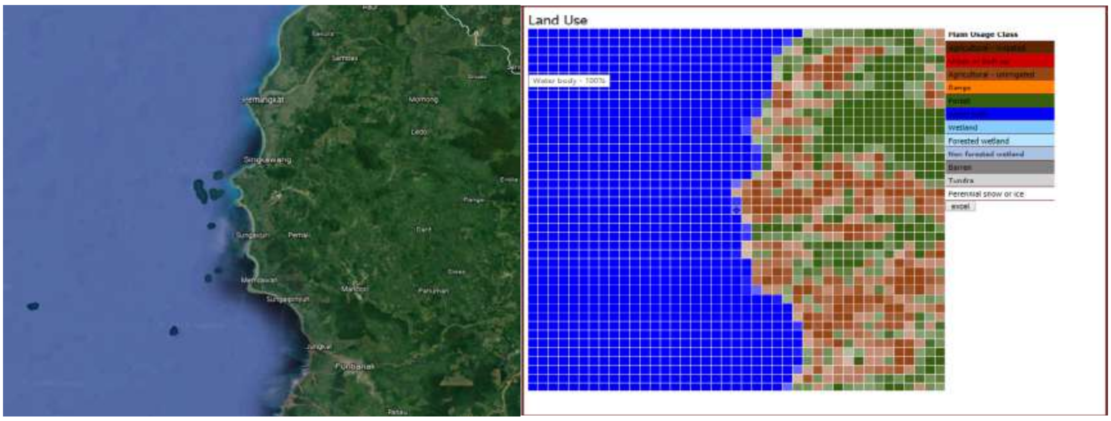


FIGURE 8. Map of land use in the area of the study

RESULTS AND DISCUSSION

The I-131 distribution is shown in **FIGURE 9** for the study area domain. Calculation results using SIMPACTS show that the spread of I-131 emissions is dominated east, south, and north. The maximum value of concentration occurs in the area around the nuclear power plant that is equal to 1.79 x 10^{-2} Bq/m^3. The area of distribution of concentration of I-131 to the east of the NPP reaches up to 50 km, the north reaches up to 80 km, and the southern reaches up to 45 km.

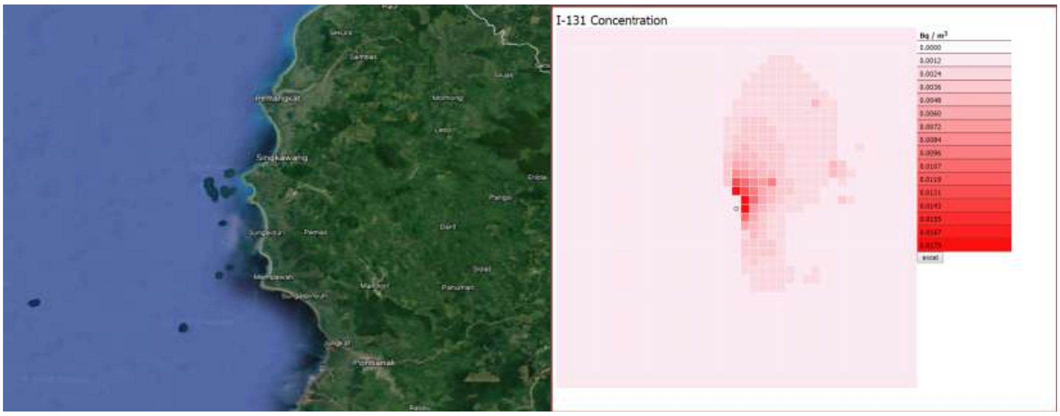


FIGURE 9. Distribution of Radionuclide I-131 distribution

The I-133 distribution is shown in **FIGURE 10** for the domain of the study area. The results of calculations using SIMPACTS show that the spread of emissions in the form of I-133 is dominated eastward, northward, and southward. The maximum value of I-133 concentration occurs in the area around the nuclear power plant that is equal to 2.8 x 10^{-2} Bq/m^3. The area of distribution of concentration of I-133 to the east of the nuclear power plant reaches up to 45 km, the north reaches up to 70 km and the southern reaches up to 35 km.

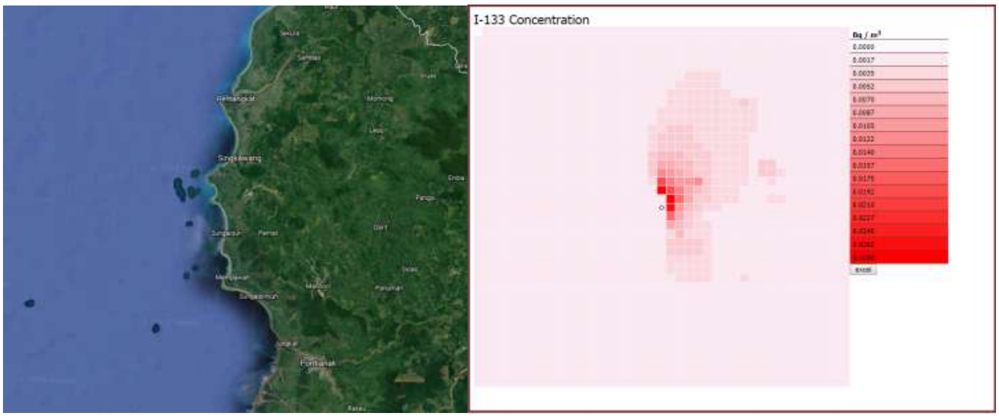


FIGURE 10. Distribution of Radionuclide I-133 distribution

The distribution of the Kr-85 distribution is shown in **FIGURE 11** for the study area domain. The results of calculations using SIMPACTS show that the spread of Kr-85 emissions is predominantly eastward, northward, and southward. The maximum value of Kr-85 concentration occurs in the area around the nuclear power plant that is equal to 1.72x10^{-3} Bq/m^3. The Kr-85 concentration distribution area to the east of the nuclear power plant reaches up to 85 km, north to 100 km, and 100 km south

FIGURE 11. Distribution of KR-85 Radionuclide distribution

The distribution of Xe-133 distribution is shown in **FIGURE 12** for the study area domain. The results of calculations using SIMPACTS show that the spread of Xe-133 emissions is predominantly eastward with an almost uniform distribution between north and south. The maximum value of Xe-133 concentration occurs in the area around the nuclear power plant that is equal to 3.15 x 10^{-2} Bq/m^3. The area of Xe-133 concentration distribution to the east of the nuclear power plant reaches up to 90 km, the north reaches up to 100 km and the southern reaches up to 95 km.

FIGURE 12. Distribution of Radionuclide Xe - 133 distribution

All radionuclide dispersion data is shown in table 3 along with the maximum concentration of radionuclides. Negative impact due to the spread of radionuclides as a result of fission, the radionuclide concentration value must be converted into units of mSv / year to find out if the value is still below the value of natural radiation. This becomes important to show that the nuclear power plant that will be built does not have the potential to interfere with human and animal health due to its radiation potential.

TABLE 3. Radionuclide Concentration Data and dispersion.

No	Type of Radionuclide	Maximum Concentration Bq/m^3	North (km)	East (km)	South (km)	West (km)
1	I-131	0,0179	80	50	45	< 5
2	I-133	0,0280	70	45	35	<5
3	Kr-85	0,00172	100	85	100	<5
4	Xe-133	0,0315	100	90	95	<5

CONCLUSION

The potential for the spread of Radionuclides from nuclear power plants which is assumed to be built in West Kalimantan precisely in the Gosong coastal area of the Bengkayang Regency has been calculated with SIMPACT software. The simulation results show that the spread is dominated towards east, north, and south. Radionuclides consisting of I-131; I-133; Kr-85 and Xe-133 are generally spread eastward to a maximum of 90 km, to the north a maximum of 100 km or more and to the south a maximum of 100 km or more. Spread to the west is very minimal because it is influenced by the direction of the wind, temperature, and air pressure which results in the spread of radionuclides further eastward. Radionuclide concentrations from NPP in West Kalimantan are still very low compared to the threshold set in government regulations for acceptable environment doses. Thus, from the point of view of radionuclide dispersion, it shows that NPP in West Kalimantan is feasible to build.

ACKNOWLEDGEMENT

Thank you to Dr. Suparman who has supported the improvement of HR competencies through the IAEA SIMPACTS E-learning training. The writing of this paper was funded by PKSEN's INEO book preparation activities

REFERENCES

[1] Birikorang SA, Abrefah RG, Sogbadji RBM, Nyarko BJB, Fletcher JJ, Akaho EHK. Ground deposition assessment of radionuclides following a hypothetical release from Ghana Research Reactor-1 (GHARR-1) using atmospheric dispersion model. *Prog Nucl Energy*. 2015;79:96-103. doi:https://doi.org/10.1016/j.pnucene.2014.11.013

[2] Vasyanovich M, Vasilyev A, Ekidin A, Kapustin I, Kryshev A. Special monitoring results for determination of radionuclide composition of Russian NPP atmospheric releases. *Nucl Eng Technol*. 2019;51(4):1176-1179. doi:https://doi.org/10.1016/j.net.2019.02.010

[3] Medved' I, Černý R. Modeling of radionuclide transport in porous media: A review of recent studies. *J Nucl Mater*. 2019;526:151765. doi:https://doi.org/10.1016/j.jnucmat.2019.151765

[4] Athar YH, Faghihi F. Dose dispersion map using the fall-out stack model of the HYSPLIT code for a pool-type 5 MW research reactor under normal operation. *Radiat Phys Chem*. 2019;165:108412. doi:https://doi.org/10.1016/j.radphyschem.2019.108412

[5] Hirose K. Atmospheric effects of Fukushima nuclear accident: A review from a sight of atmospheric monitoring. *J Environ Radioact*. 2020;218:106240. doi:https://doi.org/10.1016/j.jenvrad.2020.106240

[6] Tang Z, Cai J, Li Q, Zhao J, Li X, Yang Y. The regional scale atmospheric dispersion analysis and environmental radiation impacts assessment for the hypothetical accident in Haiyang nuclear power plant. *Prog Nucl Energy*. 2020;125:103362. doi:https://doi.org/10.1016/j.pnucene.2020.103362

[7] Pirouzmand A, Dehghani P, Hadad K, Nematollahi M. Dose assessment of radionuclides dispersion from Bushehr nuclear power plant stack under normal operation and accident conditions. *Int J Hydrogen Energy*. 2015;40(44):15198-15205. doi:https://doi.org/10.1016/j.ijhydene.2015.06.043

[8] Mohammed Saeed IM, Mohammed Saleh MA, Hashim S, bin Ramli AT, Al-Shatri SHH. Atmospheric dispersion modeling and radiological safety assessment for expected operation of Baiji nuclear power plant potential site. *Ann Nucl Energy*. 2019;127:156-164. doi:https://doi.org/10.1016/j.anucene.2018.11.045

[9] Suh K-S, Park K, Min B-I, Kim S, Yang B-M. Estimation of the radionuclides emission region using trajectory and atmospheric dispersion models. *Ann Nucl Energy*. 2016;94:626-632. doi:https://doi.org/10.1016/j.anucene.2016.03.028

[10] Li X, Xiong W, Hu X, et al. An accurate and ultrafast method for estimating three-dimensional radiological dose rate fields from arbitrary atmospheric radionuclide distributions. *Atmos Environ*. 2019;199:143-154. doi:https://doi.org/10.1016/j.atmosenv.2018.11.001

[11] Jang G, Suh S, Kim S, Suh Y, Park J. A proactive alarm reduction method and its human factors validation test for a main control room for SMART. *Ann Nucl Energy*. 2013;51:125-134. doi:https://doi.org/10.1016/j.anucene.2012.07.035

[12] Jeon J, Kim YS, Choi W, Kim SJ. Identification of hydrogen flammability in steam generator compartment

of OPR1000 using MELCOR and CFX codes. *Nucl Eng Technol*. 2019;51(8):1939-1950. doi:https://doi.org/10.1016/j.net.2019.06.024
[13] Galushin S, Kudinov P. Sensitivity and uncertainty analysis of the vessel lower head failure mode and melt release conditions in Nordic BWR using MELCOR code. *Ann Nucl Energy*. 2020;135:106976. doi:https://doi.org/10.1016/j.anucene.2019.106976
[14] Roach BD, Fenske EK, Glasgow DC, Partridge JD, Keever TJ, Giaquinto JM. Rapid concentration and isotopic measurements of ultra-trace 235U fission products with comparison to an ORIGEN isotope depletion model. *Talanta*. 2019;205:120079. doi:https://doi.org/10.1016/j.talanta.2019.06.079
[15] Pontillon Y, Geiger E, Le Gall C, et al. Fission products and nuclear fuel behaviour under severe accident conditions part 1: Main lessons learnt from the first VERDON test. *J Nucl Mater*. 2017;495:363-384. doi:https://doi.org/10.1016/j.jnucmat.2017.08.021
[16] Kim J-T, Woo S-C, Kim J-Y, Kim T-W. Debris dispersion analysis for the determination of impact conditions via traceback technology. *Int J Impact Eng*. 2018;122:472-487. doi:https://doi.org/10.1016/j.ijimpeng.2018.09.012
[17] Sunny Lim K-S, Lim J-M, Lee J, Shin HH. Impact of boundary layer simulation on predicting radioactive pollutant dispersion: a case study for HANARO Research Reactor using the WRF-MMIF-CALPUFF modeling system. *Nucl Eng Technol*. Published online 2020. doi:https://doi.org/10.1016/j.net.2020.06.011
[18] Verbruggen A, Laes E. Sustainability assessment of nuclear power: Discourse analysis of IAEA and IPCC frameworks. *Environ Sci Policy*. 2015;51:170-180. doi:https://doi.org/10.1016/j.envsci.2015.04.011
[19] Herranz M, Idoeta R, Rozas S, Legarda F. Analysis of the use of the IAEA rapid method of 89Sr and 90Sr in milk for environmental monitoring. *J Environ Radioact*. 2017;177:48-57. doi:https://doi.org/10.1016/j.jenvrad.2017.06.003
[20] Merk R. Numerical modeling of the radionuclide water pathway with HYDRUS and comparison with the IAEA model of SR 44. *J Environ Radioact*. 2012;105:60-69. doi:https://doi.org/10.1016/j.jenvrad.2011.10.014

New Paradigm for Radiation Supervisors Functional Position on Regulation Drafting Activities: A Suggestion

Fery Putrawan Cusmanri[1, a)] and Widi Laksmono[2,b)]

[1,2]*Badan Pengawas Tenaga Nuklir*

[a)]email: f.putrawan@bapeten.go.id
[b)]email: w.laksmono@bapeten.go.id

Abstract. Among the regulatory duties mandated by Act Number 10 Year 1997 concerning Nuclear Power is the making of regulations. From the perspective of human resources, functional positions that have the expertise to carry out these tasks are the radiation supervisor functional positions. Regulations regarding the items of radiation supervisory activities are regulated in Minister of Utilization of State Civil Apparatus and Bureaucratic Reform Regulation (MenPAN-RB Regulation) Number 46 Year 2012 concerning Functional Position of Radiation Supervisory Functions and its Credit Score. However, in applying these regulations, especially in the field of regulation-making, there were several problems encountered. These problems include the style of language that tends to be complex, the proportion of activities between levels that are not proportional, there are parts of the business process that are not listed, the existence of activities with unclear implementation, and similarity with the functional position of the regulation drafter. This paper is made to provide a deeper analysis of existing problems by providing solutions to each of these problems. From the results of literature studies on international standards and regulations on other functional positions that have recently been published, some recommendation that can be considered are: use terms that are easy to understand, make activities between levels more proportional based on two models: involving each level at each stage or grading for each level of difficulty, include some activities such as input from users for regulation evaluation process, remove the unclear activity implementation such as preparation of a standard manuscript, and focus on determining the matters that must be regulated in regulations related to the use of nuclear power.

Keywords: Radiation Supervisor, Functional Position, Human Resources.

INTRODUCTION

Based on Article 1 number 11 of Act Number 5 Year 2014 concerning State Civil Apparatus, functional positions are a group of positions that contain functions and duties related to functional services based on specific expertise and skills [1]. For the nuclear sector, functional positions that have functions and duties related to safety are the radiation supervisor functional positions. In line with Act Number 10 Year 1997 concerning Nuclear Energy, according to Article 8 of the MenPAN-RB Regulation Number 46 Year 2012 concerning Functional Position of Radiation Supervisory Functions and its Credit Score, among the functional activities of the radiation supervisor are radiation supervising, with sub-elements include [2]:

1. Inspection;
2. Licensing;
3. Evaluation of nuclear supervising standard norms / nuclear supervising agreements or ratification of international agreements; and
4. Certification and validation.

Among the duties of the regulatory body as stated in Article 4 of Act Number 10 of 1997 concerning Nuclear Power is regulation drafting [3], in which this is represented in the sub-element evaluation of nuclear supervising standard norms / nuclear supervising agreements or ratification of international agreements. This sub-element contains regulation drafting activities including academic manuscript/conception, legal drafting, regulation evaluation, etc. Regulatory activities themselves are carried out in the department of the Directorate of Nuclear Installation and Material Regulation (DP2IBN) and the Directorate of Radiation Facilities and Radioactive Sources Regulation (DP2FRZR) [4].

In practice, there are several issues in the implementation of the sub-element evaluation of nuclear supervising standard norms / nuclear supervising agreements or ratification of international agreements of the functional position of the radiation supervisor in the field of regulation including the style of language that tends to be complex, the elements of activity at each level are less proportional, there are activities that should be there but are not listed in the points of activity contained on MenPAN-RB Regulation Number 46 Year 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score, the existence of activities which are unclear in their implementation, and the similarity with the functional position of the regulation drafter.

The International Atomic Energy Agency (IAEA) as an international institution that issues various nuclear utilization standards has a standard publishing mechanism called the Strategies and Processes for the Establishment of the IAEA Safety Standards / SPESS. The latest version of SPESS itself was published in 2015. So it is necessary to examine the suitability of the job description for the radiation supervisor functional officer in the process of compiling the evaluation of nuclear supervising standard norms / nuclear supervising agreements or ratification of international agreements.

This paper was written to provide information on the problems that exist in the application of elements of the functional activities of radiation supervisors in the field of regulation in MenPAN-RB Regulation Number 46 Year 2012 concerning Functional Position of the Radiation Supervisor and its Credit Score and their potential for development to be applicable and in line with international standards. The writing of this paper was carried out by examining international standards, regulations regarding functional positions that were published recently and conducting studies on regulations related to the functional positions of radiation supervisors in the field of regulation. The results of the study and recommendations in this paper are also expected to be an advisor in the process of the revision of MenPAN-RB Regulation Number 46 Year 2012 in the future.

THEORETICAL BASIS

Strategies and Processes for the Establishment of the IAEA Safety Standards

In general, the preparation of standards carried out by the IAEA is as follows [5]:

1. Drafting of document preparation profile (DPP);
2. DPP approval by the authorities;
3. Drafting of draft standards (DS);
4. DS approval by the authorities;
5. Collection of input from member countries; dan
6. Issuance of safety standards.

Thus, the products produced from the above stages are DPP, DS, and safety standards (with each DPP and DS making there are review and revision stages).

In short, the DPP is a document that briefly explains the importance of a standard to be made or revised. In its description, the DPP contains input provided by member countries regarding the application of a standard, gap analysis, and theoretical foundations in making or revising a standard. In the DPP there is also a time frame for completing a draft standard starting from the preparation of the DPP until when the standard will be published along with the resources needed.

A draft safety (DS) is a standard that is drafted but not yet officially valid. Typically, the DS that is compiled will be issued and distributed officially by the IAEA to get responses from member countries within a certain period. After adjustments and considerations, the DS will then be reviewed and processed into standards that are officially published by the IAEA.

The Process of Establishing Nuclear Regulations in Indonesia

Nationally, the regulation regarding the development of regulation is regulated in Act Number 12 Year 2011 concerning Development of Regulation which then amended with Act Number 15 Year 2019 concerning Amendments to Act Number 12 Year 2011 concerning Development of Regulation.

Based on Act Number 12 Year 2011 concerning Development of Regulation, the stages of forming regulations are sequentially: planning, drafting, discussion, ratification or stipulation, and promulgation [6].

Furthermore, in Bapeten Regulation No. 8 Year 2018 concerning Procedures for Regulation Development in the Nuclear Energy Regulatory Agency, in making legislation in the environment of Bapeten an additional stage after promulgation is given, namely the evaluation of regulation. According to Article 29 paragraph (1) of Bapeten Regulation No. 8 Year 2018 concerning Regulation Development in the Nuclear Energy Regulatory Agency, it is explained that the purpose of a regulation evaluation is to perform the regulatory impact assessment [7]. The results of this evaluation can then be used as a replacement material or amendment to the regulations that have been issued.

We can briefly state that the process of regulation development in the nuclear sector is as follows:

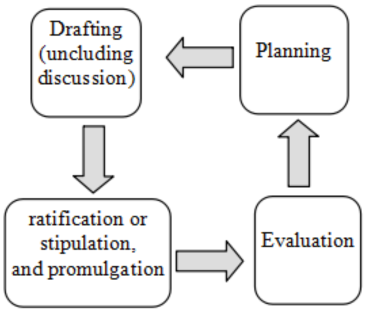


FIGURE 1. Regulation Development Life Cycle

Activities Elements of the Functional Position in Legal Drafter

One of the problems encountered in the application of MenPAN-RB Regulation Number 46 Year 2012 concerning the Functional Position of Radiation Supervisor and its Credit Score is the finding of similarities in some of the functional activities of the radiation supervisor in sub-elements of norm evaluation of nuclear supervision standards / nuclear supervision agreements or ratification of international agreements with elements Formation of Laws and Regulations in the MenPAN-RB Regulation No. 6 Year 2016 concerning Second Amendment to the Decree of the State Minister for Administrative Reform No. 41 / KEP / M.PAN / 12/2000 concerning Functional Position of the Legal Drafter and its Credit Score for functional positions of the Legal Drafter. This similarity is mainly at the stage of drafting regulations (which in PermenPAN-RB Number 46 of 2012 concerning Functional Position of Radiation Oversight and Credit Numbers are named as evaluation of nuclear supervising standard norms / nuclear supervising agreements or ratification of international agreements) with sub-elements of the regulation drafting that is owned by a functional position of the legal drafter. In general, the process carried out can be said to be similar, starting from gathering materials, compiling, discussing, and refining the draft. What distinguishes the two functional positions is the existence of grading at various levels of difficulty of the draft regulation, where the higher the level of difficulty, the functional position level responsible for an activity will also be higher [8].

Activities Elements of Functional Activities of Radiation Supervisors in the Field of Regulation

In general, based on the products produced by radiation supervisors in the DP2FRZR and DP2IBN departments, the stages of radiation supervisory in the regulatory sector sequentially (including discussion) consist of [2]:

1. drafting a proposal;
2. drafting of academic manuscript/ conceptions;
3. drafting of the standard manuscript;
4. drafting of a standard draft; and
5. drafting of standard.

The above stages generally apply to the new regulations to be issued and to be revised. A slightly different thing is that for the regulation to be revised there is an evaluation stage.

RESULTS AND DISCUSSION

From the results of the study, it can be concluded that the problems that exist in the sub-elements of functional position activities in the field of regulation include:

1. Language styles are complicated so they tend to make reading uncomfortable;
2. Less proportional between one level and another level, where there are levels which have a very small number of activities while at other levels there are very many;
3. Activities that have added value to business processes but cannot be assessed for credit score;
4. There are activities that are not implemented or it is unclear how to implement them;
5. The similarity of the process with the functional position of the draft legislator, and
6. Not aligned with the process of establishing international standards and national regulations related to the establishment of regulations.

Language Style

The use of terms is an important component in the delivery of information. The limited ability of human memory makes the use of a complex set of words at risk of increasing the complexity of information so that it is prone to make misperceptions.

The use of the term "Evaluation Of Nuclear Supervising Standard Norms / Nuclear Supervising Agreements or Ratification of International Agreements" as a sub-element title is more or less in conflict with the activities that exist in these sub-elements which mostly actually talk on the process of "drafting" and not "evaluation" [2]

The use of the term "Evaluation Of Nuclear Supervising Standard Norms / Nuclear Supervising Agreements or Ratification of International Agreements" itself is considered too long if examining the final product to be produced is basically the same, namely the provisions regarding the use of nuclear power, whether it is produced with or without being based on the ratification of the treaty international. In addition, the use of the term "standard" itself can potentially conflict with the main tasks and functions of the National Standardization Agency (BSN) [9].

The use of easy-to-understand terms like "Nuclear Power Utilization Provisions" or "Nuclear Power Provisions" can be used to make it easier for users to remember and understand the sub-elements of this activity.

Proportion Between Levels

The imbalance in the number of activities at each level can be seen from the involvement of each level at the process stage. This imbalance is particularly seen in the first radiation supervisor level which is only involved in the stages of drafting proposals, academic manuscript/ conceptions, and drafting of the standard manuscript. This involvement was only limited to the collection of materials at each stage and was not involved in the discussion.

The involvement of each level should be equitable with the optimal level of empowerment. Strategies that can be used in increasing the activity of each level can be done with two models: involving each level at each stage or grading for each level of difficulty.

Involving each level in various activities applied in functional positions in Environmental Impact Controller, where each level has an active role in almost every activity carried out by functional officials of Environmental Impact Controller [10]. This model also applied in the functional position of Curriculum Developer [11] and the functional position of Trading Supervisor [12]

For models that provide grading for each of these difficulties are applied in the functional position of Regulation Drafter[8], the functional position of Pharmacy and Food Supervisors[13], the functional position of Environmental Supervisors [14], and functional position of Quality Controller [15] which each of them divides difficulty level into various levels. The higher level of functional positions will get a more difficult job.

There Are Parts Of Business Processes That Are Not Listed

In the cycle of making regulations, the evaluation stage plays an important role to determine the level of ability of regulation and the importance of a regulation to be updated following the latest social and technological developments.

The evaluation stage has actually been alluded to in MenPAN-RB Regulation Number 46 Year 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score, but this stage only appears on the item amendment/revision [2]. In fact, regulatory evaluation activities should appear not limited to when there are amendments or revisions but are carried out periodically. The implementation of the statutory evaluation in the field of nuclear itself is required to be evaluated every 5 years [7]. The results of evaluating the application of the regulations themselves can actually vary including the replacement of regulations, changes, or actually not done anything to these regulations.

Another issue from the evaluation phase in MenPAN-RB Regulation Number 46 Year 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score is an incomplete item of activity. In the regulation evaluation activities are limited to the preparation of evaluation reports, reference collection continued with compilation, discussion, and so on until finalization [2]. In the analysis of the ability of a regulation, input from the user is absolutely necessary. Because the user is the first party to feel the benefits or difficulties in implementing a policy. However, this is not listed in MenPAN-RB Number 46 of 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score so it needs to be considered for inclusion in the revision.

Uncertain Activity Implementation

Among the items of radiation supervisor activities in the process of regulation development is the preparation of a standard manuscript. The preparation of this standard manuscript itself is positioned after the preparation of academic manuscript/ conceptions and before the preparation of the standard draft. In its application, the existence of this standard text is actually very confusing. There is still no consensus as to what the actual form of this standard manuscript is. Then, if it refers to the IAEA standard drafting process and the process of forming legislation in Indonesia, this stage is also unknown. So it is strongly recommended not to re-include (remove) this process in the revision of MenPAN-RB Regulation Number 46 of 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score.

The similarity of Process to Legal Drafter Functional Position

The similarity of the process with the functional position of legal drafter is contained in the sub-element of drafting the regulation in MenPAN-RB Regulation Number 6 Year 2016 concerning Second Amendment to the Decree of the State Minister for Utilization of State Apparatus Number 41 / KEP / M.PAN / 12/2000 concerning Functional position of Legal Drafter and its Credit Score.

Originally, in the preparation of a statutory regulation required technical input in determining the things that needed to be regulated. Based on Act Number 11 Year 2012 concerning the Development of Regulation technical input including scientific research is required in the stage of the academic manuscript [6]. For this reason, to avoid colliding with existing activities in the functional position of the regulation drafter, radiation supervisor activities in the field of regulation must focus on determining the matters that must be regulated in regulations related to the use of nuclear power. Whereas other matters such as harmonization can be left to other related functional positions.

CONCLUSION

From the results of the discussion, it can be concluded that there are still several things that are still a weakness in MenPAN-RB Regulation Number 46 of 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score. Some of these include:

1. Language style;
2. The proportion between levels;
3. There are parts of business processes that are not listed;
4. Unclear activity implementation; and
5. The similarity of the process to legal drafter functional position.

Therefore there are some recommendations which can be implemented in the revision of MenPAN-RB Regulation Number 46 of 2012 concerning the Functional Position of the Radiation Supervisor and its Credit Score:

1. Use a common term that is easy to understand;
2. Make activities between levels more proportional based on two models: involving each level at each stage mor grading for each level of difficulty;
3. Include some activities such as input from users for regulation evaluation process;
4. Remove the unclear activity implementation such as preparation of a standard manuscript; and
5. Focus on determining the matters that must be regulated in regulations related to the use of nuclear power.

ACKNOWLEDGMENTS

The author would like to thank Mrs. Dahlia Cakrawati Sinaga as Deputy Chairman for Nuclear Safety Assessment, Mr. Bambang Eko Aryadi as caretaker director of DP2IBN, as well as DP2IBN personnels for their encouragement.

REFERENCES

[1] Republik Indonesia, "Undang-undang Republik Indonesia No.5 tahun 2014 tentang Aparatur Sipil Negara," pp. 1–104, 2014.

[2] Kementerian Pemberdayaan Aparatur Negara dan Reformasi Birokrasi, "Peraturan Menteri Pemberdayaan Aparatur Negara dan Reformasi Birokrasi Nomor 46 Tahun 2012 tentang Jabatan Fungsional Pengawas Radiasi dan Angka Kreditnya," vol. 2008, no. 1, pp. 1–2, 2012.

[3] Republik Indonesia, Undang-Undang Nomor 10 Tahun 1997 tentang Ketenaganukliran, vol. 21, no. 3. Indonesia, 1997, pp. 295–316.
[4] Badan Pengawas Tenaga Nuklir, Keputusan Kepala Badan Pengawas Tenaga Nuklir Nomor 01 Rev.2/K-OTK/V – 04 tentang Organisasi dan Tata Kerja Badan Pengawas Tenaga Nuklir. 2004.
[5] International Atomic Energy Agency, "Manual for the review process for draft Nuclear Security Series Publications and for the application of SPESS A for the IAEA Safety Standards," 2015.
[6] Republik Indonesia, Undang-Undang Nomor 12 Tahun 2011 tentang Pembentukan Peraturan Perundang-Undangan. 2011.
[7] Badan Pengawas Tenaga Nuklir, Peraturan Badan Pengawas Tenaga Nuklir Nomor 8 Tahun 2018 tentang Tata Cara Pembentukan Peraturan Perundang-Undangan di Badan Pengawas Tenaga Nuklir. 2018.
[8] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 6 tahun 2016 tentang Perubahan Kedua Atas Keputusan Menteri Negara Pendayagunaan Aparatur Negara Nomor 41/KEP/M.PAN/12/2000 tentang Jabatan Fungsional perancang Peraturan Perund. 2016.
[9] Republik Indonesia, “Undang-Undang Nomor 20 Tahun 2014 tentang Standardisasi dan Penilaian Kesesuaian,” 2014.
[10] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, “Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 30 Tahun 2019 tentang Jabatan Fungsional Pengendali Dampak Lingkungan,” 2019.
[11] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, “Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 57 Tahun 2020 tentang Jabatan Fungsional Pengembang Kurikulum,” 2020
[12] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, “Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 2 Tahun 2020 tentang Jabatan Fungsional Pengawas Farmasi dan Makanan,” 2020.
[13] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, “Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 45 Tahun 2020 tentang Jabatan Fungsional Pengawas Perdagangan,” 2020
[14] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, “Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 22 Tahun 2019 tentang Jabatan Fungsional Pengawas Lingkungan Hidup,” 2019.
[15] Kementerian Pendayagunaan Aparatur Negara dan Reformasi Birokrasi, “Peraturan Menteri Pendayagunaan Aparatur Negara dan Reformasi Birokrasi Nomor 54 Tahun 2020 tentang Jabatan Fungsional Penguji Mutu Barang,” 2020

This page is intentionally left blank…

2020 ANNUAL
NUCLEAR SAFETY SEMINAR

HELD BY
IN COLLABORATION WITH

P2STPIBN-BAPETEN
Jl. Gajah Mada No. 8
Jakarta

FMIPA UNIVERSITAS INDONESIA
UI Campus
Depok

@bapeten
www.bapeten.go.id

@fmipa_ui
www.sci.ui.ac.id