KnE Energy

ICoNETS Conference Proceedings
International Conference on Nuclear Energy Technologies and Sciences (2015), Volume 2016

Knowledge E
Engaging minds

Conference Paper

Evaluation of Radionuclides Release Estimation of Power Reactor using Scdap/Relap

Jupiter Sitorus Pane, Surip Widodo

Center for Nuclear Reactor Technology and Safety, National Nuclear Energy Agency of Indonesia (BATAN), Puspiptek Area Building No. 80, Serpong, Tangerang Selatan 15310, Indonesia

Abstract

Incident of radiation release to the environment is important event in reactor safety analysis. Numerous studies have been conducted using various computer codes, including SCDAP/RELAP, to calculate radionuclide releases into the reactor coolant during severe accident. This paper contains description of calculation results of radionuclide release from reactor core to primary coolant system in a 1000 MW PWR reactor with the aim to study behavior of radionuclide releases during severe accident. The calculations using SCDAP/RELAP was done by assuming that there has been a station black out which ends up with some vapor released into the containment. As a result, the water level in core was reduced up to a level where the core is no longer covered by water. The uncovered core heats up to certain temperature where the oxidation of the cladding started to occur. Afterwards the oxidation generated heat made fuel melting temperature reached and as consequences the release of radionuclide to the primary coolant. The calculations show that in parallel with the increasing of fuel temperature, the radio nuclide releases into the gap through diffusion started at time of 2000 seconds after initial simulation but with a neglected concentration. Subsequently at the time of 29200 seconds, the temperature reached more than 1000 K and the oxidation of the Zr-cladding material occurred which accelerated the fuel temperature increase and as well as radionuclide release. At 34000 seconds, maximum temperature of core reached 2800 K and radionuclide release into the primary cooling system started. At this time, accumulated dissolve fission product reached amount of 74.5 kg, while the non-condensable radionuclide reached 122 kg. However, these value need to be investigated further.

Keywords: SCDAP/RELAP, radionuclide release, severe accident

Corresponding Author:
Jupiter Sitorus Pane,
Surip Widodo,
email: jupiter_pane@batan.go.id

Received: 29 July 2016
Accepted: 21 August 2016
Publishing: 15 September 2016

Publishing services provided by Knowledge E

Selection and Peer-review under the responsibility of the ICoNETS Conference Committee.

1. Introduction

Radionuclide release of fission products into the environment is an important event for estimating the magnitude of impact from construction of nuclear installation at a particular site. This analysis is part of the Safety Analysis Report preparation. The amount of releases of radionuclide into the environment is determined by the quantity of inventory, releases of radionuclide into primary coolant system and confinement. SCADAP/RELAP developed by INEEL provides calculation method to estimate such release into primary system during severe accident [1,2,3,4].

Generally, an estimation of radionuclide releases is performed using removable fraction of radionuclide released to containment as described in NUREG 1465 [5,6]. A simple method of source term calculation has also been studied in order to obtain the relationship of radionuclide releases fraction to the temperature of the primary cooling system due to the heat up by residual heat and oxidation of the cladding material Zirconium (Zr) [7,8]. By using the code of SCDAP/RELAP one can study the mechanism of accident starting from reactor shutdown due to outage of electricity supply (Station Blackout), the increase of coolant and fuel temperature, cladding oxidation as well as heat up acceleration until the melting of fuel. As a consequences, the release of radionuclide into the gap between fuel matrix and cladding and to coolant occurred. The output of the code include fuel fission product inventory, fuel rod gap inventory, release of non-condensable gas to coolant, release of soluble fission product to coolant and fission product transport [9].

In this paper the author would like to describe the results of calculation of radionuclide releases to primary cooling system using SCADAP/RELAP code with the aim to investigate the behavior of radionuclide releases during severe accident conditions and to evaluate the capacity of SCDAP/RELAP in doing such calculation. As a case study, the author uses1000 MW Surry PWR as the Power Reactor model.

The evaluation of radionuclide release was done by assuming that there was a station blackout at the PWR Surry and the reactor was immediately shutdown and all active safety systems failed to operate. Due to residual heat in the core, the temperature of primary coolant increased as well as its pressure. The power operated relief valve (PORV) in the Pressurizer (PZR) was set for nominal operation at 2335 psig and at 2385 psig for high pressure trip and at 2485 psig for safety valve [10]. The fluctuation of the pressure in primary system made the PORV open and close and as a result the steam generated in the primary cooling system released. This led to emptying the Pressurizer Vessel as well as the Core Vessel and called as loss of coolant [11]. This loss of coolant caused the increase of fuel temperature up to its melting point so that radionuclide could release.

An evaluation using SCDAP/RELAP to estimate inventory fission product and releases of radionuclide into the gap and primary cooling system has been performed by using an experimental device namely BOIL-OFF and by applying simple nodalization [12]. Indeed, the fuel core was modeled as one component and one node. In this analysis, the suitability of the calculation result using SCDAP/RELAP, by comparing to FASTGRASS or PARAGRASS and ORIGEN II, has been proven. However, study on source term calculation upon RINGHALS 2 PWR using SCDAP/RELAP shows an inconsistent result where calculation of the accumulated long term non-condensable radionuclide release to the coolant have huge different compared to the transport routine [13]. Therefore, in addition to evaluate the radionuclide release behavior, this study also identifies the capability of SCDAP/RELAP code in estimating radionuclide release to the coolant.

2. Theory

Accident Mechanism

In the event of Station Blackout i.e. outside power supply outages totally, the entire reactor safety function does not work completely. Although the reactor in a shutdown state as a result of reactor safety protection respond to an accident, the reactor coolant still receives heat from the fuel heat residual that causes the coolant temperature rise even reaching the melting point of the fuel if no further action taken to stop the heating. Measures to prevent the continuation of the incidents to a more severe level referred to as the act of managing accidents or also called accident management. The increase in temperature and pressure in the primary coolant can cause the increase of pressure in PORV valve that can activate PORV valve to open. In contrast, the opening of the PORV valve causes the decreaseof primary pressure up to a certain pressure limit that brings PORV back to close. These conditions, opening and closing the valve, caused the decreasing of coolant in the vessel and eventually can cause the fuel become uncovered. This condition triggers the increasing of heating up of the primary system even reaching the temperatures above 1000 K, which causes oxidation of the cladding material Zirconium (Zr). This oxidation will produce hydrogen and heat. An accumulation of residual heat and additional oxidation heat will accelerate the fuel temperature increases even it could reach the melting point of the fuel. In a solid material like fuel and cladding, the higher the temperature the faster the diffusion occurs.

Diffusion of Fission Products

In general, the fission products created in the fuel element matrix diffused in the fuel matrix through diffusion model. Especially for Xenon, Krypton, Cesium, Iodine and Tellurium the mathematical model was shown in a diffusion equation as follows [14].

$$\frac{\partial C_g}{\partial t} = \frac{1}{r^2}\frac{\partial}{\partial r}\left[D_g r \frac{\partial C_g}{\partial r}\right] + K_g \left[\frac{\partial C_g}{\partial t} = \frac{1}{r^2}\frac{\partial}{\partial r}\left[D_g r \frac{\partial C_g}{\partial r}\right] + K_g\right] \quad (1)$$

With,

C_g = concentration of fission product gases (fission / m3)
D_g = diffusion coefficient of atoms
K_g= atom production rate
r = radius of the fuel (m)

For less volatile fission products, the radionuclide release is modeled using CORSOR-M model. In this model the assumed rate of release at the spots for each species is formulated as [15]:

$$FP = FFP \cdot \left(1 - e^{(-FRC(I) \cdot DTIME)}\right) \quad (2)$$

with,

FFP = mass of the species present on the stain at the beginning of time step (Kg)
FRC = coefficient release rate
FRC value calculated by the Arrhenius equation:

$$FRC = KO(I) \cdot e^{\left(\frac{-Q(I)}{1.987 \cdot 10^{-3}\, T}\right)} \quad (3)$$

with,

KO (I) and Q (I) = constant release of the species with the values in Tabel 1 [15].

Transfer of Radionuclide of Fission Products to the Fuel and Cladding Gap

Diffusion of fission products move towards the edge of the fuel and gap. Rate and magnitude of movement is strongly influenced by temperature, oxidation, interaction with components of the cladding and the structure of burn-up, fuel type and morphology. Fission product gases released from the fuel is assumed to reach the surface of the fuel with migration successively from the surface of the particle (grain face) to the particle surface and ends at the edge of the particle. This model can be applied to gases derived from fission, bubble nucleation and resolution, migration of bubbles, bubble coalescence, gas bubbles and the formation of the channel on the surface of the particle (grain face), interlinked porosity on grain edges, grain boundary micro cracking and grain growth and grain boundary sweeping.

Table 1: Value KO(I) and Q(I).

Species	KO(min^{-1})	Q(kcal/mol)
UO2	1,46E7	143,1
Zr	2,67E8	188,2
Sn(clad)	5,95E3	70.8
Fe	2,94E4	87
Ru	1,62 E6	152,8
Zr(clad)	8,55E4	139,5
Ba	2,95E5	100,2
Sr	4,40 E5	117,0
Te	2,00E5	63,8
Ag	7,90E3	61,4
Cs*	2,00E5	63,8
I*	2,00E5	63,9

3. Methodology

The study begins with developing nodalization of core and primary system of 1,000 MW PWR reactor Surry, develop programs and inputs in accordance with the conditions and parameters of the reactor, analyze and evaluate the results of the calculation of the maximum temperature of the core and the estimated fission products in accordance with the development of the maximum temperature of the core. Table 2 shows the relevant data in the simulation calculations of Surry PWR radionuclide releases.

Table 2: Data Related to Surry PWR in Simulation SBO.

Parameters	Value
Power (MW)	2443
Pressure (MPa)	14,7
Operation Temperature (K)	500
Fuel element configuration	157 (15 x 15)
Active fuel element height (m)	3,66
Accumulated borated water mass (kg)	29100 (at T = 322 K)
Accumulator initial pressure (MPa)	4,24
Fuel rod number	240
Fuel pellet stack length (m)	3,6576
Control rod	21
Fuel pellet stack (m)	4.634
Inconnel spacer grid location	0,0; 1,46; 1.83; 2,19; 2,93

DOI 10.18502/ken.v1i1.468

Since the object of this study is similar to previous studies [11, 12] then nodalization, parameters value and the program calculation is also the same but with different subject. The previous studies dealt with performance of the primary system evaluation of Surry PWR but now dealt with radionuclide releases of fission product.

Nodalization of core was figure out by dividing the fuel into 5 components named component number 1, 3, 5, 7 and 9and controls into 5 component named component number 2, 4, 6, 8, 10. Each component is divided into 10 nodes. Component 1, 3, 5, 7 and 9, each consist of 1020, 4080, 7344, 12240, 7334 rod. For the simulation, it is assumed that there has been failure on PORV valve during station black out accident. The system shutdown 100 seconds after the accident that make power decrease until 10% power.

4. Results and Discussion

The Increase in Fuel Temperature

In this simulation, the station black out accident assumed to be occurred. The reactor was shut down at 100 second after the accident so that the power reduced up to 10% from the full power. This residual heat could be accumulated reaching a very high fuel temperature if no additional cooling system operated. Figure 1 shows the increase in the maximum fuel temperature in the reactor core.

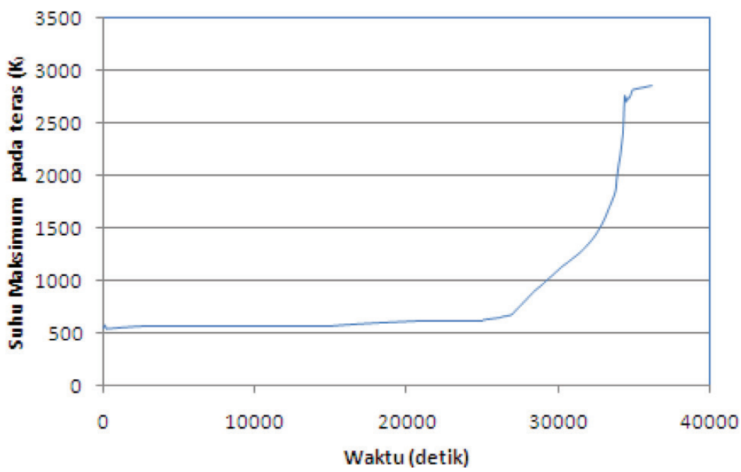

Figure 1: The maximum temperature rise of 1,000 MW reactor core.

The increase of temperature became significant starting from a few seconds to the 27200 second where the core became uncovered by the cooling and the fuel temperature reached 722 K. In parallel with the declining water levels in the core, the fuel temperature continues to rise until it reaches the temperatures above 1000 K in the second of 29200. At this temperature, Zr fuel cladding oxidation started to occur.

The Mechanism of Oxidation of the Cladding

Oxidationbetween vapor and Zr cladding material at temperature of 1000 K would generate heat and hydrogen gas as shown in Figure 2 and Figure 3. In these both pictures, the amount of hydrogen and heat rise due to oxidation events. The heat generated by this oxidation is accelerating fuel temperature where at the second of 34400 the fuel temperature reaches 2768 K. At this point, the heat coming from the oxidation begins to decline but still give heat

DOI 10.18502/ken.v1i1.468

accumulation and the maximum temperature of the fuel keep rising to 2868 K. With this temperature the fuel reach its melting pointl. At this temperature SCDAP /RELAP can no longer continue the calculation.

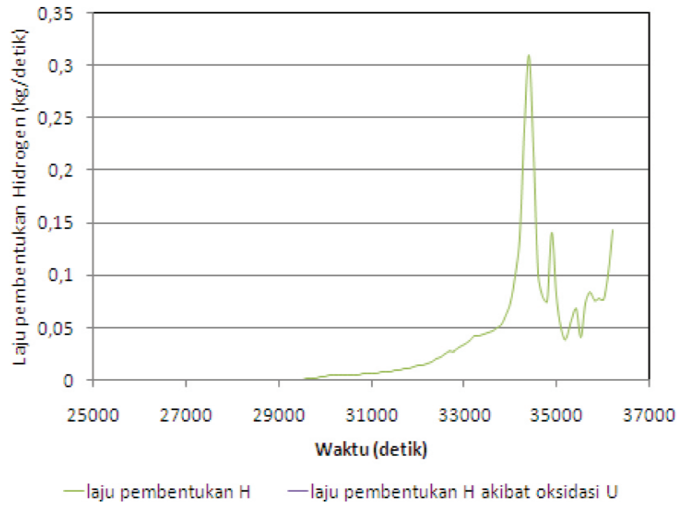

Figure 2: The rate of hydrogen formation during oxidation.

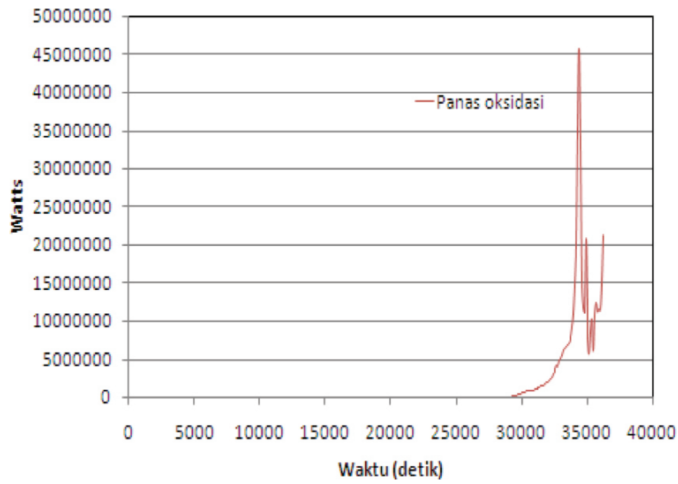

Figure 3: The heat generated during the production of Hydrogen.

Inventory of Fission Products

The fission products were formed during the fission process. Number of fission products is strongly determined by its previous reactor operating history. The code of SCDAP/RELAP provide the inventory of the core based on its operating history. Table 3 shows the fission product inventory at the nodes and components 1, 3, 5, 7 and 9 of PWR 1000 MW fuel core in the form of gas (Xe, Kr) and volatile (Cs, I, Te) with a history of being operated with a power density 3.6138e8 W / m3 within 5.9205E+7 seconds.

Release of Radionuclide and Core Damage

The inventory generated in the fuel matrix diffused with the increase of fuel temperature. From the simulation results it was detected that the diffusion output is visible from the second of 2000 with a very small of amount, for example at component 1 the diffusion of radionuclide

DOI 10.18502/ken.v1i1.468

to the gap was about 1.79 E-18 kg and therefore it cannot be detected. This number will always be the same until it reached the seconds of 32000 in where the temperature reached the value of 1356 K. Indeed, the number of radionuclide in gap approaching 3.1 E-09 Kg.

Table 3: Inventory of gas and volatile fission products.

Node	Xe (kg)	Kr (kg)	Cs (kg)	I (kg)	Te (kg)
1	5.9450E-04	6.6958E-05	3.4416E-04	2.3987E-05	5.5231E-05
2	8.1511E-04	9.1805E-05	4.7187E-04	3.2888E-05	7.5727E-05
3	1.0731E-03	1.2086E-04	6.2123E-04	4.3298E-05	9.9695E-05
4	1.0651E-03	1.1996E-04	6.1660E-04	4.2975E-05	9.8953E-05
5	1.0348E-03	1.1655E-04	5.9905E-04	4.1751E-05	9.6135E-05
6	1.0186E-03	1.1472E-04	5.8965E-04	4.1097E-05	9.4628E-05
7	1.0140E-03	1.1421E-04	5.8702E-04	4.0913E-05	9.4205E-05
8	1.0116E-03	1.1393E-04	5.8561E-04	4.0815E-05	9.3980E-05
9	9.8024E-04	1.1040E-04	5.6747E-04	3.9551E-05	9.1068E-05
10	7.9064E-04	8.9049E-05	4.5771E-04	3.1901E-05	7.3453E-05

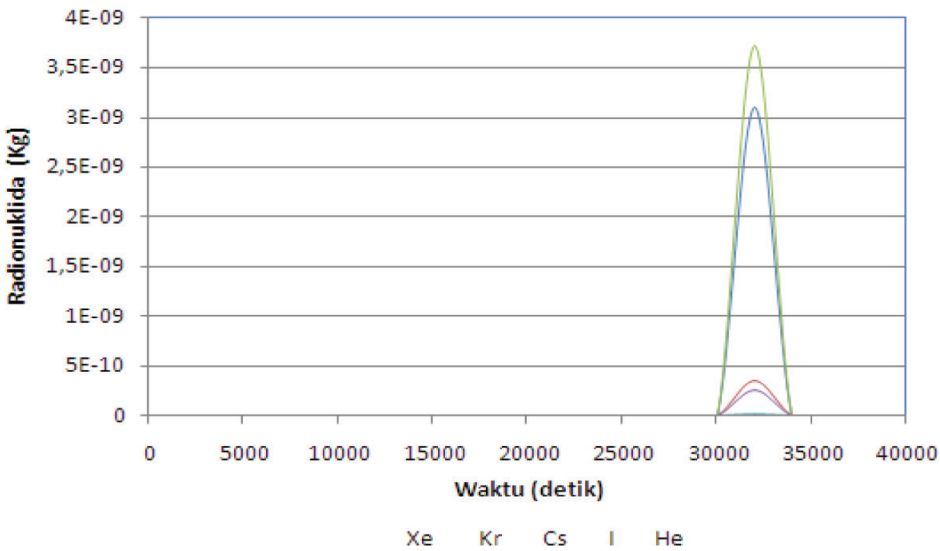

Figure 4: Removable radionuclide into fuel and cladding gap in component 1.

By increasing the temperature up to 2112 K in the second of 34000 then radionuclide not just stay in the gap but also escape to the coolant. In the second of 36200 in where temperatures reaching 2868 K, radionuclide material that reached gap will entirely migrate to cooling system with Xe and Kr are around 122 kg and CsI and CsOH of 74.5 Kg. In these conditions most of the cladding material has melted [16]. In the chart, the release of radionuclide to the cooling can be described as in Figure 5. However these numbers are out of the expected number where the values are much more than the total inventory product in gap. This condition was also identified in during performing source term calculation for RINGHALS 2 PWR [13]. This fact needs to be studied further. It is hypothesized that the problem arise due to the accumulated error of calculation, not because of the model. The longer the time, the bigger the deviations.

Nevertheless, from this analysis, it can be summarized that releases of radionuclide to the cooling depend on the size of the initial inventory from the beginning of reactor operation, type of accident, time and temperature since the accident occurred. The release of radionuclide can be stopped when the reactor can be controlled before the temperature reaches the melting temperature of the fuel cladding by equipping reactor with its protection system. Therefore, this computer code is very useful for accident managements analysis.

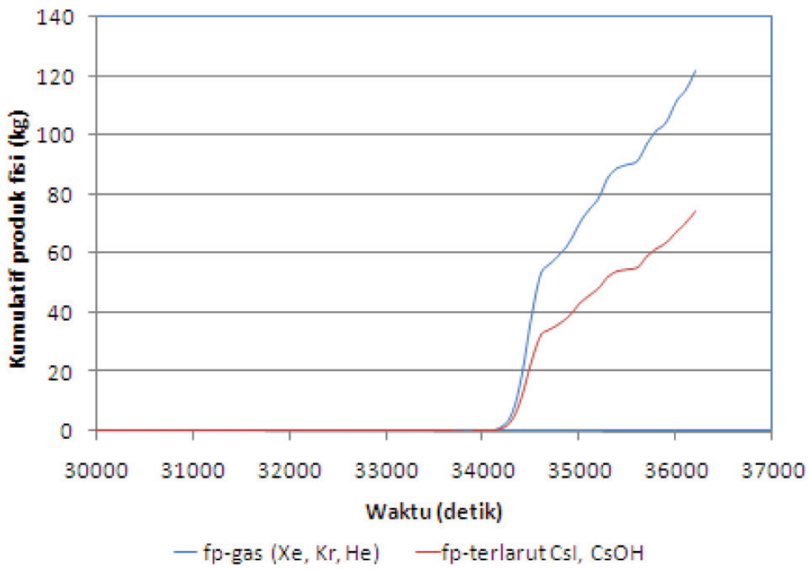

Figure 5: Radionuclide release to the primary coolant of 1000 MW reactor.

5. Conclusion

The behavior of radionuclide release of 1000 MW during severe accident highly depend on initial inventory of the core and heat up process during the period of accident. The radionuclide releases into the gap and proceed into the primary cooling system begins when the temperature has reached above 1356 K i.e. after the oxidation of Zr cladding material and hydrogen production occurred. The increase of radionuclide was rocketed when most of the cladding material melts. This release phenomenon can be used as a reference in designing a reactor accident management system of 1000 MW to mitigate releases of radionuclide into the coolant.

Further study to develop SCDAP/RELAP code capability need to be performed since the calculation result of radionuclide release to the primary cooling system of non condensable species are not as expected. It is hypothesized that the problem arise due to the accumulated error of calculation, not because of the model.

References

[1] MEHBOOB, K., XINRONG, C., Source Term Evaluation Of Two Loop PWR Under Hypothetical Severe Accidents" Annals of Nuclear Energy, Volume 50, December 2012, *Pages 271-284*

[2] KIM, S. G., GYUNO, Y, and SEONG, P. H., Prediction Of Severe Accident Occurrence Time Using Support Vector Machines, Science Direct, Journal Elsevier, Nuclear Engineering and Technology, 2015.

[3] HASTE T. J., BIRCHLEY J., RICHNER M., Accident Management Following Loss-Of-Coolant Accidents During Cooldown In A Westinghouse Two-Loop PWR, Nuclear Engineering and Design, Volume 240, 2010, Pages 1599-1605

[4] Urbonavičius E., Kaliatka A., VileiniškisV. Integral analysis of large LOCA beyond design basis accident scenario in RBMK-1500, Nuclear Engineering and Design, Volume 240, Issue 3, March 2010, Pages 616-629

[5] SOFFER L., BUSRON S. B., FERREL C. M., LEE R. Y., Accident Source Terms for Light-Water Nuclear Power Plants, U. S. Nuclear Regulatory Commission, NUREG 1465, Washington, DC, 1995.

[6] Mehboob, K., Xinrong, C., Source term evaluation of two loop PWR under hypothetical severe accidents, Annals of Nuclear Energy, Volume 50, December 2012, Pages 271-284

[7] Shi X, Cao, X, Liu, Z., Oxidation behavior analysis of cladding during severe accidents with combined codes for Qinshan Phase II Nuclear Power Plant, *Annals of Nuclear Energy, Volume 58, August 2013, Pages 246-254*

[8] Wang J., Tian, W, Fan, Y., Mao, K., Lu, J., Su, G., Qiu, S., The development of a zirconium oxidation calculating program module for Module In-vessel Degraded Analysis Code MIDAC, *Progress in Nuclear Energy, Volume 73, May 2014, Pages 162-171*

[9] SCDAP/RELAP5-3D Code Developer Team, SCDAP/RELAP5-3D Code Manual, Assessment of Modelling of Reactor Core Behaviour During Severe Accidents, INEEL/EXT-02-00589, Revision 2-2, 2003.

[10] VEPCO, Surry Power Station Unit 1 and 2, Technical Specification, DOCKET NOS. 50-280 AND 50-281, Virginia, USA, 1991,

[11] PANE J. S. An Evaluation of Primary System of PWR Reactor during Station Blackout Accident Using SCDAP/RELAP (Evaluasi Kinerja Sistem Primer Reaktor PWR Pada Kejadian *Station Black Out* Menggunakan Relap/Scdap), Prosiding Seminar SENTEN, Kalimantan Barat, 2014

[12] PANE J. S., Pemodelan Dan Simulasi Lepasan Radionuklida Pada Perangkat Percobaan *Boil-Off* Dengan Scadap/Relap, dipresentasikan pada Seminar Nasional Teknologi Energi Nuklir 2014, 2014.

[13] JOHANSSON, LL, Source Term Calculation-Ringhals 2 PWR, Studsvik Eco & Safety AP Nykoping, Sweden, 1998.

[14] HIDAKA, A., SODA, K., SUGIMOTO, J., " SCDAP/RELAP5 Analysis of Station Blackout with Pump Seal LOCA in Surry Plant, Journal of Nuclear Science and Technology, 2012.

[15] BERA, B., KUMAR, M., THANGAMANI, I., PRASAD, H., SRIVASTAVA, A., MAJUMDAR, P., DUTTA, A., VERMA V., GANJU S., CHATTERJEE B., LELE H.G., RAO, V.V.S.S.. GHOSH, A.K. "Estimation of source term and related consequences for some postulated severe accident scenarios of Indian PHWR" Nuclear Engineering and Design, An International Journal devoted to all aspect of Nuclear Fission Energy, Elsevier, 2010.

[16] Park, R.J., KANG, K.H., Hong, S.W. Hong, Kim HY, Detailed Evaluation of Melt Pool Configuration In The Lower Plenum Of The APR1400 Reactor Vessel During Severe Accidents, Annals of Nuclear Energy, Volume 75, January 2015, Pages 476-482