Nucl. Sci. and Tech, Vol.4, No. 1 (2014), pp. 10-25

Nuclear Science and Technology

Journal homepage: https://jnst.vn/index.php/nst

Design Analyses for Full Core Conversion of The Dalat Nuclear Research Reactor

Luong Ba Vien, Le Vinh Vinh, Huynh Ton Nghiem, Nguyen Kien Cuong

Reactor Center – Nuclear Research Institute – Vietnam Atomic Energy Institute

01 Nguyen Tu Luc, Dalat, Lamdong

Email: reactor@hcm.vnn.vn

Abstract: The paper presents calculated results of neutronics, steady state thermal hydraulics and transient/accidents analyses for full core conversion from High Enriched Uranium (HEU) to Low Enriched Uranium (LEU) of the Dalat Nuclear Research Reactor (DNRR). In this work, the characteristics of working core using 92 LEU fuel assemblies and 12 beryllium rods were investigated by using many computer codes including MCNP, REBUS, VARI3D for neutronics, PLTEMP3.8 for steady state thermal hydraulics, RELAP/MOD3.2 for transient analyses and ORIGEN, MACCS2 for maximum hypothetical accident (MHA). Moreover, in neutronics calculation, neutron flux, power distribution, peaking factor, burn up distribution, feedback reactivity coefficients and kinetics parameters of the working core were calculated. In addition, cladding temperature, coolant temperature and ONB margin were estimated in steady state thermal hydraulics investigation. The working core was also analyzed under initiating events of uncontrolled withdrawal of a control rod, cooling pump failure, earthquake and MHA. Obtained results show that DNRR loaded with LEU fuel has all safety features as HEU and mixed HEU-LEU fuel cores and meets requirements in utilization as well.

Keywords: *HEU, LEU, neutronics, thermal hydraulics, safety analyses*

I. INTRODUCTION

In this full core conversion study, neutronics, thermal hydraulics and safety analysis were carried out to investigate characteristics of LEU working core fully loaded with LEU fuel. All computer codes were validated with HEU and mixed cores.

Using MCNP [6], REBUS-PC [5] and VARI3D computer codes, a series of static reactor physics calculation were performed to obtain neutronics parameters of the working core (see **Fig. 1**). Some parameters included in the design of working core with shutdown margin, excess reactivity taking into account of irradiated Beryllium poisoning, control rod worths, detailed power peaking factors, neutron performance at the irradiation positions, reactivity feedback coefficients, and kinetics parameters. Because the higher content of ^{235}U in a LEU FA compared to HEU FA, it is needed to rearrange the fuel assemblies and berrylium rods with the different way to the first HEU core to meet the safety requirements.

Thermal hydraulics parameters at steady state condition were obtained by using PLTEMP3.8 code [11] introduced models and correlations that suitable for the concentric tube fuel type and natural convection regime of the DNRR.

Based on the neutronics analysis parameters of the LEU core, the postulated

https://doi.org/10.53747/jnst.v4i1.209

Received 5 March 2014, accepted 10 March 2014

transients and accidents selected for the DNRR are analyzed. The RELAP5/MOD3.2 code [15] was used for analysis of RIA (Reactivity Initiated Accident), LOFA (Loss Of Flow Accident) transients.

These study results showed that a LEU core loaded with 92 fuel assemblies and 12 beryllium rods around the neutron trap satisfies the safety requirements while maintaining the utilization possibility similar to that of the previous HEU and recent mixed fuel cores.

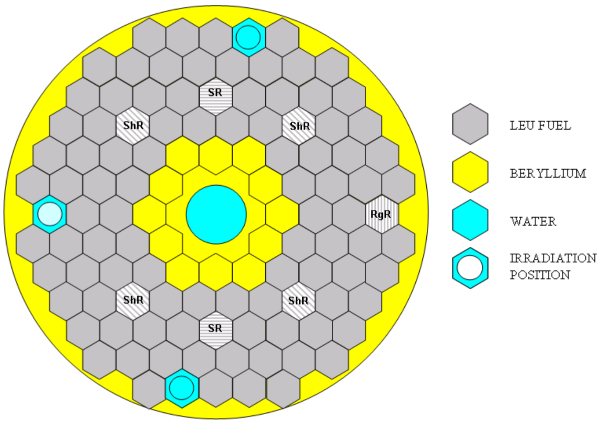

Fig. 1. The new designed working core loaded with 92 LEU FA and 12 Beryllium rods.

II. CALCULATION MODELS AND COMPUTER CODES

A. Neutronics and Thermal Hydraulics Calculation

The diffusion code REBUS-PC with finite difference flux solution method was used to perform core calculation for reactor physics characteristics and operation cycle calculations with micro neutron cross sections according to 7 energy groups (collapsed from 69 energy groups) that were generated by WIMS-ANL code [4]. The FA cross sections were generated in a radial geometry with each fuel element depleted based upon its unique neutron spectrum in the WIMS-ANL model. The REBUS-PC fuel depletion chains included production of six Pu isotopes, Am-241, Np-237, and lumped fission product. Isotopic precursors of Xe-135 and Sm-149 were also included in the depletion chains so that Xe and Sm transients during periods of shutdown and startup could be modelled.

REBUS-MCNP Linkage [7] was used to calculate burnup distribution using "two way" linking option in which MCNP is used for calculating neutron flux and cross section in one group neutron energy and burn up calculation is implemented by REBUS-PC.

The MCNP5 code using an ENDF-B/VI cross section library was used to construct a detailed geometrical model of each reactor component and calculate control rod worths, multiplication coefficient, power distribution, neutron flux performance in irradiation positions, reactivity feedback coefficients, and kinetics parameters (prompt neutron life time and delayed neutron fraction).

A detailed geometrical model of reactor components including all fuel assemblies, control rods, irradiation positions, beryllium and graphite reflectors, horizontal beam tubes and thermal column was made in the MCNP model, except in the axial reflectors above and below the fuel assembly where some materials were homogenized. **Fig. 2a** provides the radial and axial models of the reactor for Monte Carlo Calculations.

The kinetics parameters were calculated also by VARI3D code. The real and adjoint fluxes which are required to compute these parameters were provided by DIF3D-a main module of REBUS-PC code.

In diffusion theory, the reactor was modeled in hexagonal geometry with a heterogeneous representation of the fuelled and non-fueled portions (see **Fig. 2a**). Each homogenized fuel assembly was modelled using five equal volume axial depletion zones. The beam tubes were modeled using a homogenized mixture of air or concrete, graphite and aluminum.

The reactor models for diffusion and Monte-Carlos computer codes were validated by comparing with good agreement not only to the fresh HEU configuration cores but also to the HEU burnt cores. These models were then applied for partial core conversion analyses of DNRR [3]. The measured data collected during the deployment of partial core conversion project showed that the predicted calculation results are quite acceptable [8,9].

The PLTEMP/ANL3.8 [15] thermal-hydraulics code for plate and concentric-tube geometries with capability of calculating natural circulation flow was used for thermal-hydraulics analyses. A chimney model as well as Collier heat transfer correlation and CHF Shah's correlation have been recently implemented make the code suitable DNRR thermal-hydraulics calculation.

Fig. 2b shows the model of WWR-M2 fuel assembly, core and chimney of the DNRR for PLTEMP code. A fuel assembly was modelled as three concentric cylindrical tubes.

Before using PLTEMP code to calculate for DNRR with fully LEU fuel assemblies, the code was validated by comparing analytical results with experimental results of mixed-core.

B. Transient/Accidents analyses

The DNRR has three barriers as other research reactors that prevent or limit the transport of fission products to the environment, which are fuels and cladding, reactor pool water and reactor confinement. The safety system settings are showed in **Table I**.

Table I. Safety system settings.

Parameters	Safety system settings
Maximum thermal power (Pmax)	550 kW (110% FP)
Minimum reactor period (Tmin)	20s
Deficient level of pool water	60 cm
Primary coolant flow rate	40 m^3/h
Secondary coolant flow rate	70 m^3/h

In the Safety Analysis Report (SAR) for the DNRR [1], the possible initiating events were classified by groups. The initiating events in each group are then analyzed and justified in order to identify the limiting event that will be selected for further detail quantitative analysis. The limiting event in each group has potential consequences that exceed all others in that group. Limiting events were selected for detailed analyzed are as follows: (1) Uncontrolled withdrawal of a control rod; (2) Primary/Secondary Pump Failure; (3) Earthquake; (4) Fuel cladding failure. A summary of the core parameters used for the safety analysis is given in **Table II**.

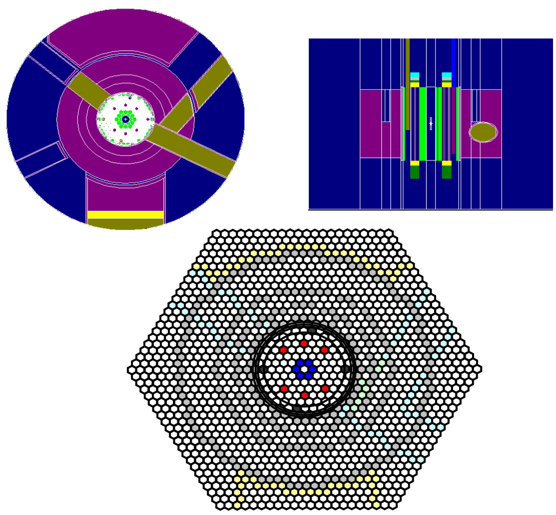

Fig. 2a. Radial and Axial models for Monte Carlo calculations (upper) and Radial model for Diffusion Theory calculations (under).

To ensure the fuel clad integrity in operational condition and to protect the public and the environment in case of accident, in the SAR for the DNRR, the following acceptance criteria were defined:

- For anticipated operational occurrences:

(1) Minimum margin to departure from nucleate boiling (DNB) shall be over 1.5;

(2) Maximum temperature of fuel cladding shall not exceed 400°C;

(3) Fuel cladding integrity shall be assured.

- For accident conditions:

(1) Core covering shall be maintained;

(2) Core shall not be remarkably damaged;

(3) Release of fission products into the environment shall not be remarkable.

The RELAP5 code was used for analyzing the events of excess reactivity insertion by uncontrolled withdrawal of a control rod and earthquake. The piping of the primary cooling system and pool volume were divided into nodes with similar dynamic characteristics. The reactor core was divided into 2 channels with axial nodes. The hot channel represents the hottest channel in the

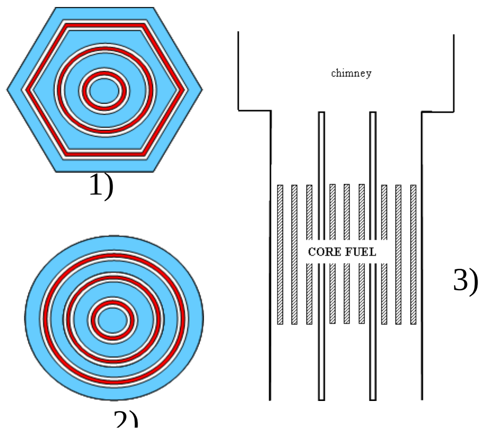

Fig. 2b. DNRR model for PLTEMP (1-fuel assembly cross-section; 2-FA model for PLTEMP; 3-reactor coolant system model).

core corresponding to a cooling channel with maximum heat flux. The average channel represents the rest of the cooling channels. Each channel was modelled as three fuel element plates and four coolant flow gaps. The nodding diagram of the DNRR for RELAP5/3.2 is presented in **Fig. 2c.**

The MACCS2 code [19] was used to estimate the radiological impact of the hypothetical accident on the surrounding public. The core radiation inventories were calculated by ORIGEN2 code [20] using neutron cross-sections of the actinides obtained from MCNP5 code.

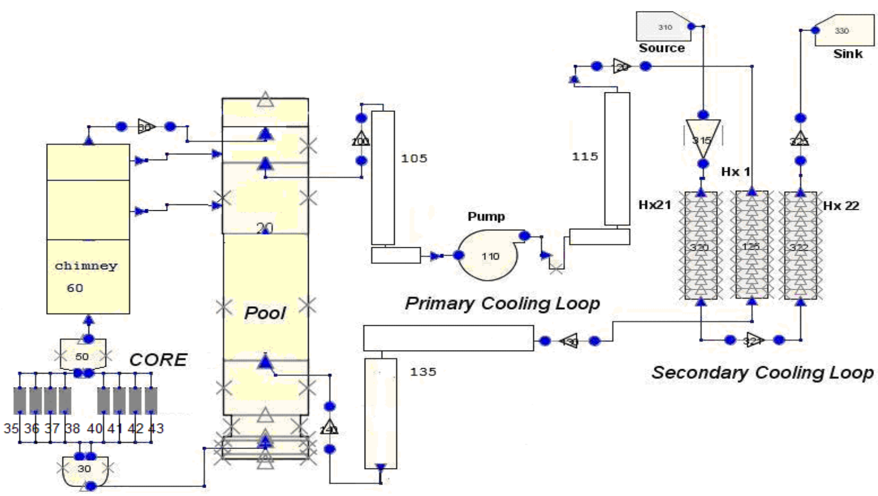

Fig. 2c. Nodding diagram of DNRR for RELAP5/3.2.

Table II. Core parameters used for safety analysis.

Parameters	Values
Power, kW	500
Coolant inlet temperature, °C	32
Peaking factor (shim rods at 300 mm)	
- Axial peaking factor	1.363
- Radial peaking factor	1.376
- Local peaking factor	1.411
Reactor kinetics	
- Prompt neutron life, s	8.925×10^{-5}
- Delayed neutron fraction (1$)	7.551×10^{-3}
Temperature reactivity coefficients	
- Moderator, %/K; (293-400°K)	$- 1.264\times10^{-2}$
- Fuel, %/°C;	
(293-400°K)	$- 1.86\times10^{-3}$
(400-500°K)	$- 1.92\times10^{-3}$
(500-600°K)	$- 1.56\times10^{-3}$
- Void, %/% of void	
(0-5%)	-0.2432
(5-10%)	-0.2731
(10-20%)	-0.3097
Reactivity control	
- Shutdown worth, % (2 safety rods)	3.7
- Maximum withdrawal speed of one shim rod, mm/s	3.4
and of the regulating rod, mm/s	20
Reactor protection characteristics	
- Response time to overpower scram, s	0.16
- Response time to fast period scram, s	
Start-up range	9.1
Working range	6.7
- Drop time of control rods, s	0.67

III. RESULTS AND DISCUSSIONS

A. Neutronics and Thermal Hydraulics

^{3}He and ^{6}Li Poisoning of Irradiated Beryllium [10]

Since 1984, the DNRR has been put into operation with a considerable amount of Beryllium used for neutron trap at the core center and periphery for improving neutron reflection around. Because Beryllium has large thermal neutron absorption cross sections, the buildup of ^{3}He, ^{6}Li and ^{3}H concentrations results in large negative reactivities which alter flux and power distributions.

Program Beryl [10] has been modified to calculate the ^{3}He, ^{6}Li and ^{3}H concentrations. The MCNP5 was then used to determine the poisoning effect of ^{3}He, ^{6}Li and ^{3}H concentrations on reactor core reactivity. The comparison of reactivities between calculation results and measured data of some beryllium blocks irradiated in DNRR (**Table III**) shows that the negative reactivity of irradiated beryllium determined by above-mentioned method is reliable. Six beryllium rods were used for measurement, two fresh beryllium rods and four irradiated beryllium rods (two beryllium rods at the end 1994 and two at the end 2002). 9-6 and 5-6 positions were chosen to measure reactivity of couple beryllium rods through changing position of control rod (Regulating Rod). The error of control rod position is estimated about 0.4 cent. Following calculation scheme for beryllium poisoning above, reactivity of the poisoning process in new configuration cores about -1$. All calculation for design LEU cores, beryllium poisoning is included in the model for MCNP code.

Table III. Comparison of calculated and measured of reactivities of irradiated beryllium rods in DNRR.

	Measured reactivity (Cent)	**Calculated reactivity (Cent)**	**Error (%)**
2 Beryllium Rods at the end 1994	-3.89 ± 0.4	-4.65 ± 0.0038	16.34
2 Beryllium Rods at the end 2002	-6.28 ± 0.4	-7.19 ± 0.0039	12.66

The working core characteristics

From the calculation results of shutdown margins, excess reactivities, power peaking factors, and neutron performance at the irradiation positions of 4 candidates cores, the working core with the better features from the safety and utilization point of view was chosen for detailed analysis. The main calculated characteristics of working core is showed in the **Table IV**. The shutdown margins of the core is met the safety requirement of -1.0%. Calculated neutron flux at the neutron trap of the core is nearly the same as that of mixed core (92HEU+12LEU). **Table V** shows the control rod worths. Detailed neutron flux performance at the main irradiation positions are presented in **Table VI**.

Table IV. Calculation results of working core compared with current mixed core.

Parameters	**LEU Core**	**Current Mixed Core**
Excess Reactivity (%) – Fresh	6.63	
Excess Reactivity (%) – After 600FPDs	3.79	
Shutdown Margin (%) – Fresh	-2.92	-4.56
Shutdown Margin (%) – After 600 FPDs	-6.62	
Radial Power Peaking Factor		
Control Rods Out	1.398	1.431
Control Rods In	1.434	
Thermal Neutron Flux at Neutron Trap Center (n/cm^2)		
Control Rods Out	2.22E+13	2.22E+13
Control Rods In	2.14E+13	
Fast Neutron Flux at Neutron Trap Center (n/cm^2)		
Control Rods Out	1.95E+12	3.15E+12
Control Rods In	1.92E+12	

Table V. Control Rods worths (%Δk/k).

Control Rods	Core1 Fresh	MCNP error	Core1 Burnt	MCNP error
Shim rod 1	2.5896	0.000091	2.3539	0.000091
Shim rod 2	2.6100	0.000111	2.4033	0.000124
Shim rod 3	2.7784	0.000118	2.5381	0.000122
Shim rod 4	2.4687	0.000122	2.2604	0.000117
Regulating rod	0.4363	0.000126	0.3629	0.000119
Safety rod 1	2.1955	0.000106	2.3084	0.000115
Safety rod 2	2.2356	0.000119	2.3579	0.000105

Table VI. Neutron flux performance.

Irradiation positions		Thermal, <0.625eV ($n/cm^2.s$)		Epithermal, <0.821MeV ($n/cm^2.s$)		Fast, <10MeV ($n/cm^2.s$)	
		Fresh	Burnt	Fresh	Burnt	Fresh	Burnt
Neutron Trap	Maximum	2.07E+13	2.20E+13	6.79E+12	7.12E+12	1.83E+12	1.92E+12
	Average	1.45E+13	1.49E+13	6.00E+12	6.04E+12	1.62E+12	1.63E+12
Channel 13-2	Maximum	9.45E+12	9.86E+12	8.19E+12	8.42E+12	2.98E+12	3.02E+12
	Average	7.00E+12	7.12E+12	6.53E+12	6.51E+12	2.46E+12	2.44E+12
Channel 7-1	Maximum	5.41E+12	5.66E+12	9.63E+12	9.76E+12	4.22E+12	4.26E+12
	Average	4.11E+12	4.18E+12	7.23E+12	7.15E+12	3.19E+12	3.15E+12
Channel 1-4	Maximum	9.24E+12	9.71E+12	8.02E+12	8.22E+12	2.92E+12	2.99E+12
	Average	6.85E+12	7.01E+12	6.41E+12	6.40E+12	2.42E+12	2.40E+12
Rotary Specimen	Average	3.55E+12	3.56E+12	7.58E+11	7.56E+11	1.93E+11	1.93E+11

Power Distribution and Power Peaking Factors

Power peaking factors of the core with different position of control rods were calculated and presented in **Table VII**. The maximum power peaking factor is in position of control rods at 250 mm. Detailed axial power distribution according to control rod position was also calculated. Radial power distributions at different control rod position are showed in **Fig. 3**.

Table VII. Power peaking factor according to control rod positions

Position (mm)	Peaking Factor			
	F.A. Radial	Core Radial	Axial	Total
0	1.378	1.398	1.296	2.498
150	1.378	1.399	1.343	2.589
200	1.375	1.403	1.356	2.615
250	1.377	1.409	1.365	2.648
300	1.376	1.411	1.363	2.646
350	1.378	1.415	1.336	2.605
600	1.378	1.434	1.284	2.537

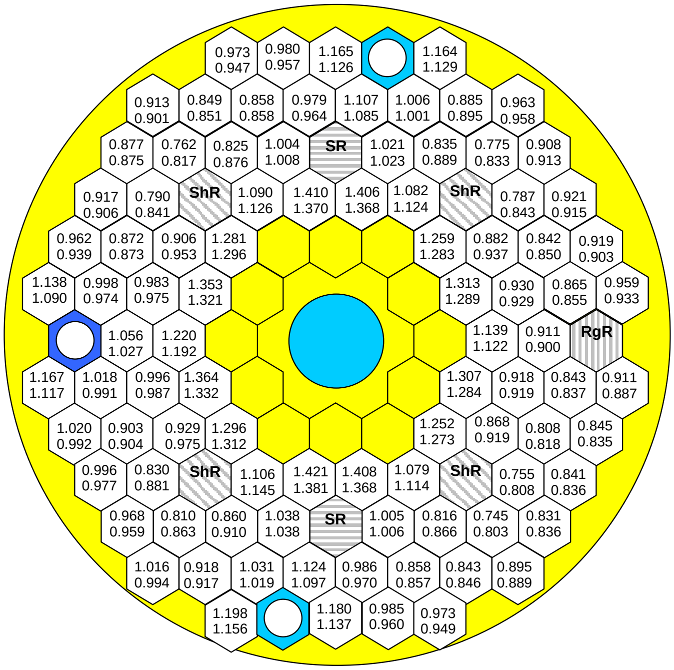

Fig. 3. Radial power distribution (Upper values: Fresh Core; Under values: Burnt Core)

Reactivity Feedback Coefficients and Kinetics Parameters

Reactivity feedback coefficients calculated with the MCNP5 are depicted in **Table VIII**. The negative results of reactivity feedback coefficients show the inherent safety of the LEU core. **Table IX** shows the kinetics parameters of the LEU cores calculated using the VARI3D and MCNP5 codes. The calculated results from the two computer codes are in good agreement. These data will be used in transient calculation for safety analysis of fully LEU core of DNRR.

Table VIII. Feedback reactivity coefficients.

Parameter	**DATA**	±σ
Moderator Temperature Reactivity Coefficient (%/°C)		
293 °K to 400 °K	-0.01317	0.00005
Fuel Temperature (Doppler) Reactivity Coefficient (%/°C)		
293 °K to 400 °K	-0.00192	0.00005
400 °K to 500 °K	-0.00182	0.00003
500 °K to 600 °K	-0.00154	0.00002
Moderator Density (Void) Reactivity Coefficient (%/% of void)		
0 to 5 %	-0.2514	0.0011
5% to 10 %	-0.2784	0.0012
10 % to 20 %	-0.3255	0.0006

Table IX. Calculated results of kinetics parameters for LEU core.

Family, i	**Decay Const.** λ_i (s^{-1})	**Relative Yield** a_i	**Fraction** β_i
1	1.334E-02	3.507E-02	2.648E-04
2	3.273E-02	1.804E-01	1.363E-03
3	1.208E-01	1.742E-01	1.315E-03
4	3.030E-01	3.843E-01	2.902E-03
5	8.503E-01	1.594E-01	1.204E-03
6	2.856E+00	6.666E-02	5.033E-04
Total delayed neutron fraction, β			
VARI3D			7.551E-03
MCNP5 – Fresh			7.761E-03
MCNP5 – Burnt			7.762E-03
Prompt neutron life time, ℓ			8.925E-05

Burn up calculation

The first cycle length was estimated by REBUS-MCNP Linkage system code. Burn up calculations were performed by assuming that shim rods and regulating rod were in critical position following each burn-up step. The value of reactivity for Xe-135 poisoning was estimated about 1.2% Δk/k. The result of depletion shows that operating time may be extended about 11 years (calculated with 1300 hours per year) or 600 full power days (FPDs). The burn up of U-235 reached average value of 8.2% and maximum value of 11.4%. In the next cycle, about 8 fuel assemblies will be inserted so the reactor core will operate with 100 fuel assemblies. The **Fig. 4** shows burn up distribution after 600 FPD operation.

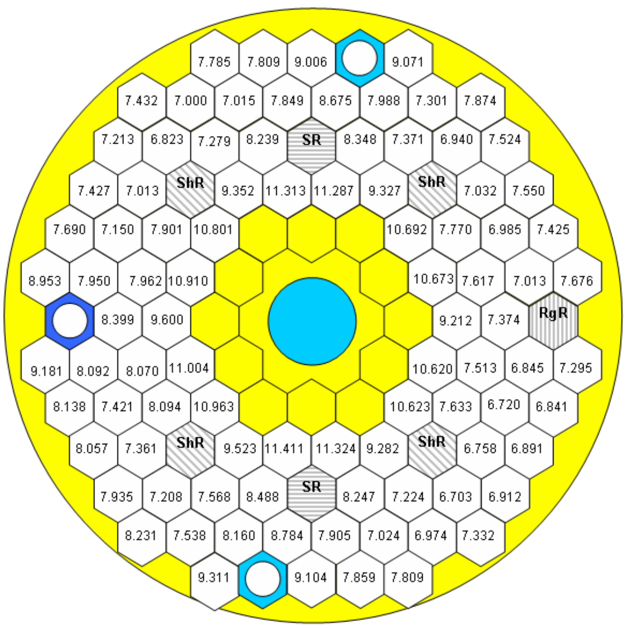

Fig. 4. Burn up distribution using REBUS-MCNP Linkage system after 600 FPD.

The PLTEMP code was used for calculating cladding temperature, coolant temperature and safety margins for the candidate cores. The calculated results are presented in **Table X** and **Fig. 5**. At nominal power without uncertainties and maximum permissible inlet temperature (32°C), the maximum cladding temperature is 90.50°C. Calculation was carried out for nominal power with systematic errors (equivalent to 70kW power) and the maximum cladding temperature is 95.69°C. In this case, by using Shah's correlation, the obtained minimum DNBR is 9.9. The minimum flow instability power ratio (MFIPR) is 2.04. From above-mentioned calculated results, it may conclude that the working core meets the requirements of thermal hydraulics safety. At the power of 500kW with systematic errors, maximum cladding temperatures are below the permissible value of 103°C [2] and far below the ONB temperature (estimated about 116°C using Forster-Greif correlation). The maximum outlet coolant temperature is calculated about 60°C, much lower than saturated temperature (108°C).

Fig. 6 shows the comparison of cladding temperature of 92FA LEU cores and 89FA fresh HEU core. Compared to the 89FA fresh HEU core established in 1984, cladding temperature of working core is about 2°C lower.

Table X. Cladding temperature and ONB margin by PLTEMP Code.

Distance (cm)	500kW				550kW		600kW	
	without sys. error		with sys. error		with sys. error		with sys. error	
	Tc(°C)	ΔT-ONB(°C)	Tc(°C)	ΔT-ONB(°C)	Tc(°C)	ΔT-ONB(°C)	Tc(°C)	ΔT-ONB(°C)
2.5	63.91	51.89	66.89	49.24	68.95	47.39	70.96	45.59
7.5	70.56	45.59	74.13	42.36	76.58	40.14	78.97	37.97
12.5	78.46	38.07	82.71	34.18	85.63	31.51	88.46	28.91
17.5	84.83	31.90	89.61	27.50	92.89	24.48	96.05	21.57
22.5	88.77	27.95	93.85	23.26	97.33	20.06	100.68	16.95
27.5	90.50	26.05	95.69	21.25	99.23	17.97	102.65	14.80
32.5	89.86	26.34	94.95	21.63	98.43	18.41	100.76	16.40
37.5	87.10	28.58	91.91	24.13	94.41	21.94	96.22	20.48
42.5	83.98	31.14	88.24	27.24	89.94	25.90	91.57	24.60
47.5	79.67	34.76	82.92	31.91	84.43	30.74	85.89	29.59
52.5	74.91	38.73	77.42	36.64	78.79	35.57	80.13	34.52
57.5	71.21	41.70	73.32	40.02	74.64	38.98	75.94	37.93

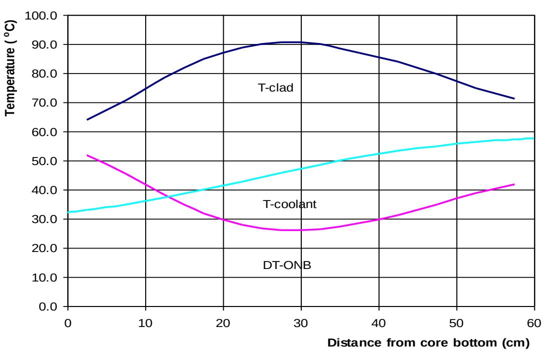

Fig. 5. T/H parameters at 500kW without uncertainties.

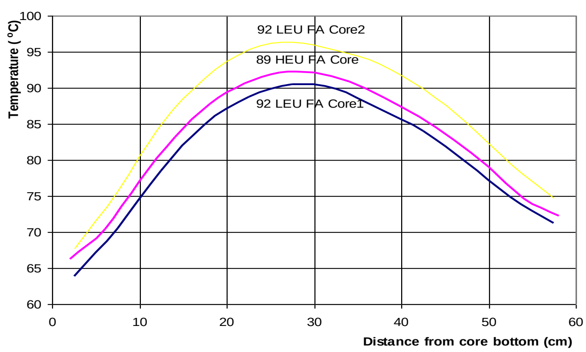

Fig. 6. Comparison of calculated cladding temperature between 92FA LEU cores and HEU core.

2. Transient/Accidents analyses

Uncontrolled withdrawal of one shim rod or the regulating rod

In this event, it is assumed that one of the shim rods or the regulating rod is withdrawn in the most effective part from 200 mm to 400 mm at the speed for 3.4 mm/s of shim rod and 20 mm/s for regulating rod. The initial conditions are as follows:

a) Start-up case:

(1) -1% $\Delta k/k$ sub-critical; Power level: 10^{-5}%FP; Coolant inlet temperature: 32^{o}C.

(2) Critical state; Power level: 10^{-3}%FP; Coolant inlet temperature: 32^{o}C.

b) Steady-state operation:

Power level: 100%FP; Coolant inlet temperature: 32^{o}C.

In sub-critical status, when one shim rod is inadvertently withdrawn with the speed of 3.4 mm/s, from the core, the reactor power only increases to the maximum value of 2.78×10^{-7} MW while the fuel cladding temperature is unchanged. With initial conditions of criticality at the power level of 10^{-3}%FP (5×10^{-6} MW) if there is no fast period signal and the overpower trip setting is 110%FP, the fuel clad temperature reaches to 97.8^{o}C, but still far below ONB (Onset of a Nucleate Boiling) temperature.

The event of one shim rod inadvertently withdrawal with speed of 3.4 mm/s from stable operation of 100%FP (500 kW) are showed in **Fig. 7** and **Table XI**. In this case, the reactor power increases and reaches to the over-power setting value (110%FP) within 3.39 seconds generating a scram signal. After a delay time of 0.16 seconds the reactor power is rapidly suppressed because of the control rods insertion. The peak power of the reactor is only attained 0.553 MW with a slight increase of the maximum fuel cladding temperature. With the assumption of no overpower scram signal appeared, a fast period scram signal is generated after 8.33 seconds from the initiation of transient event. The reactor will be shutdown after 6.7 second delay with a peak power of 0.957 MW. The maximum fuel cladding temperature is predicted to be 113.0^{o}C without any nucleate boiling occurrences. The minimum DNBR (Departure from Nucleate Boiling Ratio) estimated about 6.5 is much higher than the acceptance criterion of 1.5.

With the same initial conditions, the calculated results for the event of withdrawal of the regulating rod are slightly different from those of above-mentioned event, when one shim rod is withdrawn. This can be explained by the similar insertion rate of reactivity in the two cases (about 0.02$/s). The regulating rod has lower reactivity worth but higher withdrawal velocity compared to those of a shim rod.

Table XI. Transient results of one shim rod withdrawal from 100%FP.

Parameters	Values	
	110%FP Scram	**Period Scram**
Time to Peak Power, s	3.6	15.1
Peak Power, MW	0.553	0.957
Time to Peak Clad Temperature, s	3.7	15.2
Peak Clad Temperature, ^{o}C	91.9	113.0
Minimum DNBR		6.5

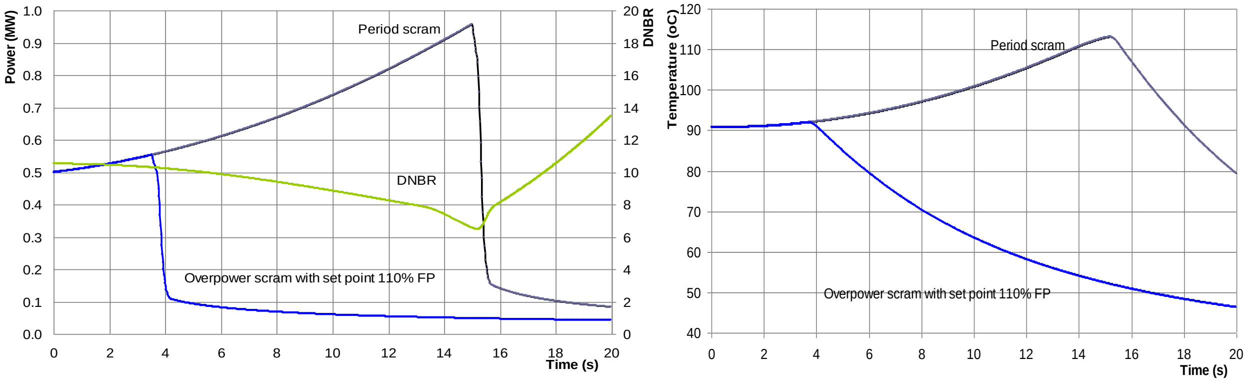

Fig. 7. Reactor power and cladding temperature transient of one shim rod withdrawal from a stable operation of 100%FP.

Cooling pump failure

In the event of in-service primary or secondary cooling pumps stopped working, the reactor is automatically shutdown by an abnormal technological signal on low flow rate (the setpoint is 40 m^3/h for the primary flow, and 70 m^3/h for the secondary flow). The residual heat after shutdown is about 6% FP (30 kW) in maximum and the natural convection process can itself assure the good cooling of the core.

If the reactor is purposely maintained at full power operation, failure of cooling pumps leads to loss of heat removal from the pool water, and thus gradually increases of the pool water temperature. The results show that the clad temperature reaches the maximum allowable operating clad temperature of 103 ^oC at about 55 min; i.e. the reactor could continue its operation for 55 minutes within the envelope of the limiting conditions of operation. The results also show that even at the end of the simulation (7000 s) the clad temperature has been well below that of the acceptance criterion for anticipated operational occurrences.

Earthquake

The postulated event of an earthquake of intensity grade VI is assumed to occur while the reactor is at full power. Owing to the measures undertaken in design and construction, the removal of all control rods would not exceed 10 mm and insert a step positive reactivity estimated of 0.3$. With this reactivity insertion, the scram set-point of reactor overpower is attained almost instantaneously. If the reactor scram is initiated by overpower signal with a delay of 0.16 sec, the fuel surface temperature increases slightly before decreases with the power, the residual heat after shutdown is sufficiently removed from the fuel by natural convection of pool water without considerable increase of the temperature.

Fig. 8 shows the analyzing results of the earthquake event assuming the protection system fails to shutdown the reactor, and Because of the loss of offsite power due to the earthquake, the primary and secondary pumps stop operating. In this case, the reactor power increases to the max value of 1.525 MW after 200 seconds from the initiation of this event. The reactor power then rapidly decreases because the significant increasing of core water temperature so that the positive reactivity insertion is overtaken by the negative reactivity feedback (about -0.44$). The reactor is then kept at subcritical state. The cladding temperature reaches a maximum value of 118.2^{o}C, then decreases with no significant overheating of the fuel. The maximum outlet water reaches 89°C and gradually decreases to a value at about 60°C, which is still far below the saturation temperature. The minimum DNBR of 4.79 is much higher than the acceptance value.

In case the cooling pumps remain working after the earthquake event (very unlikely); the peak power reaches 1.57 MW within 300 seconds and decreases due to negative temperature feedback to a stable value of about 1.12 MW. The cladding temperature reaches to a maximum value of 118.38°C then gradually decreases to a stable value of 115°C without nucleate boiling. The maximum temperature of outlet water is 89°C at the peak power then decreases and stabilizes at about 82°C, well below the saturation point. The minimum DNBR in this case estimated about 4.74 is still far from the acceptance criterion.

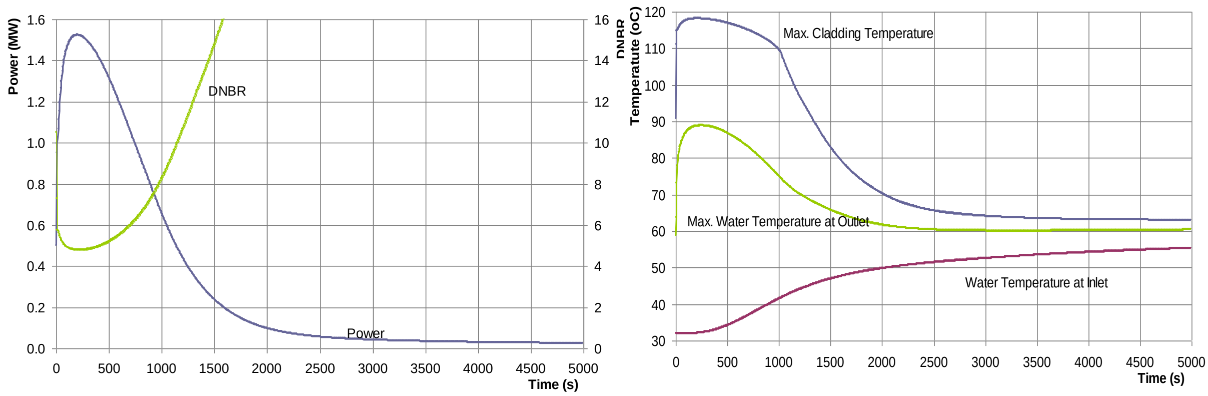

Fig. 9. Power and Temperature responses to earthquake event while cooling pumps are stopped functioning.

Fuel cladding failure (MHA)

For the derivation of source term of this event, it is assumed that no core melting occurs but cladding rupture of one fuel assembly is involved. It is also assumed that the damaged fuel assembly is irradiated at the maximum neutron flux position in the core and the fuel damage occurs immediately at the end of operating cycle of 100 hrs with no decay.

From the damaged fuel assembly, 100% of noble gases (Xe, Kr), 25% halogens (I), and 1% of other radionuclides (Cs, Te) [21] are released directly to the reactor building with the assumption of no retention of volatile fission products in the pool water. During the accident evolution, the emergency ventilation system is not in place, the normal ventilation system V1 is in operation but HEPA filter with 95% efficiency is not available, and there are no decay and deposition of radionuclides within the reactor building.

The evaluation of dose to a member of the public is calculated by code MACCS2 version 1.13.1, using the following assumptions: (1) The radionuclides are released to the environment through the 40 m stack; (2) The Gaussian plume model is used to calculate air concentration of radioactivity; (3) Tadmor and Gur parameterization is used for this analysis; (4) No building in the vicinity (an open area release), plume rise mechanics only due to momentum rise (non-buoyant plume) and no wet deposition are assumed; (5) The dry deposition velocity is assumed to be 0.01 m/s, which corresponds to a particle with an aerodynamic equivalent diameter of 2 μm to 4 μm (for unfiltered particulate releases) [15]; (6)

Surface roughness length is specified as 50 cm; (7) Mixing layer height is assumed to be 500 m (see Table 36 in Appendix VII of Ref. 21); (8) The breathing rate is 3.3×10^{-4} m^3/s; (9) No shielding and sheltering are assumed; (10) Doses at each downwind distance are calculated for one year after the arrival of the plume (11). The environmental release is assumed to begin at the start of the weather conditions: Pasquill class D2.0 (most frequent stability class and most frequent wind speed).

The effective equivalent doses, including cloudshine dose, inhalation dose and groundshine dose, as a function of the distance from the source are shown in **Table XII** and **Fig. 10**. It is seen that radiation exposure to the general public with the maximum effective dose of 0.64 mSv/year at distance from 400 m to 500 m from the stack. This value is lower than the annual dose limit of 1.0 mSv specified for the public [22].

Table XII. The annual effective dose to the public vs distance for the MHA.

Distance (m)	Effective Dose (mSv)	Distance (m)	Effective Dose (mSv)
50	4.80E-02	1100	3.18E-01
150	1.43E-01	1300	2.59E-01
250	4.95E-01	1500	2.16E-01
350	6.42E-01	1700	1.83E-01
450	6.44E-01	1900	1.57E-01
550	5.94E-01	2250	1.23E-01
650	5.33E-01	2750	9.14E-02
750	4.74E-01	3250	7.08E-02
850	4.21E-01	3750	5.66E-02
950	3.75E-01	4250	4.64E-02

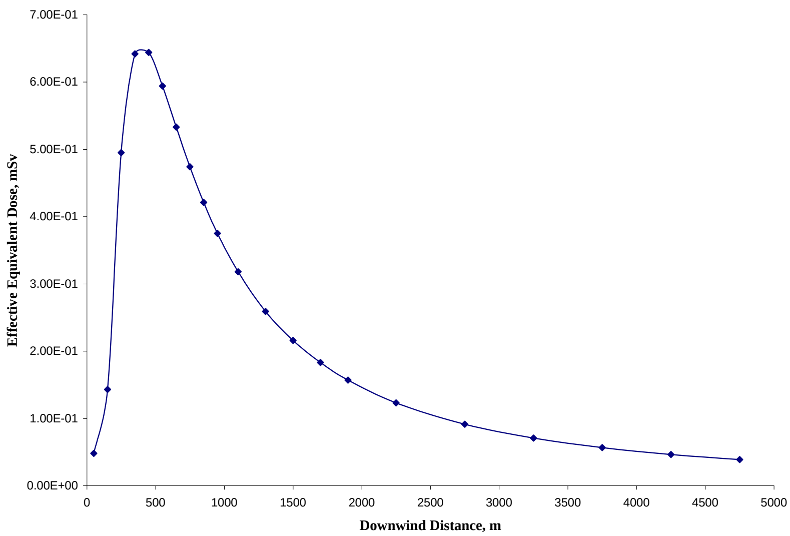

Fig. 10. The annual effective dose to the public in MHA event within 5 km.

IV. CONCLUSIONS

Neutronics, steady-state thermal-hydraulic and transient/accidents analyses for Dalat Nuclear Research Reactor show that with a slight change in arrangement of Be rods, the main features of 92 LEU WWR-M2 FA cores are equivalent to those of HEU and current mixed fuel cores.

The negative values of reactivity feedback coefficients show the inherent safety feature and shutdown margin of both candidate cores meets the safety required value of -1% Δk/k. The working core with 92 fresh LEU fuel assemblies can be operated for 600FPDs or about 11 years based on the current operating schedule without shuffling. The neutron fluxes at the irradiation positions are not much different from those of the current mixed fuel core.

In thermal hydraulics aspect, the requirement of thermal-hydraulic safety margin for two candidate cores in normal operational condition is satisfied. The calculated maximum cladding temperature in operational condition is below the permissible value of 103^{o}C.

In transient/accidents aspect, some postulated initiating events and accident related to the conversion of the DNRR to full LEU core were selected and analyzed. Based on the calculated results, conclusions might be withdrawn as following:

- The excess reactivity insertions when inadvertent withdrawals of control rod from start-up or nominal power operation are prevented by safety settings to initiate the reactor scram at overpower and fast period. None of these initiators would lead to the ONB and DNB, ensuring the integrity of the fuel cladding. The residual heat after shutdown is sufficiently removed from the fuel by natural convection of pool water.

- If one of the cooling pumps stopped working, the reactor is automatically shutdown by a scram signal on low flow rate. The decay heat is removed from the fuel by natural convection of pool water. In this event, if the reactor was purposely maintained at full power, it could be safely operated for 55 minutes when maximum cladding temperature is still lower than the permissible value of 103^{o}C.

- The postulated earthquake event of MSK intensity grade VI would cause a step reactivity insertion of 0.3$. Even if the reactor fails to be scrammed, this positive reactivity can be covered by negative temperature feedback if the cooling pumps are stopped simultaneously, keeping the reactor sub-critical. In case the cooling pumps continue operating after earthquake event, the negative temperature feedbacks act to bring the reactor power to a stable level of about 1.12 MW without nucleate boiling. The minimum DNBR is much higher than the acceptance criterion of 1.5.

- The maximum hypothetical accident assumes 100% of noble gases (Xe, Kr), 25% halogens (I), and 1% of other radio-nuclides (Cs, Te) in a most power fuel assembly after a long run are released into the environment through 40m high stack. This event is considered to be very unlikely to occur for the DNRR. Even so, it would not cause undue radiological risk to the environment or the public.

ACKNOWLEADGMENTS

The authors would like to express their gratitude to experts from the Reduced Enrichment for Research and Test Reactors (RERTR) program of Argonne National Laboratory for financial support as well as very useful discussions during design calculation of full core conversion for the Dalat Nuclear Research Reactor.

REFERENCES

[1] Safety Analysis Report for the Dalat Nuclear Research Reactor, (2003).

[2] VVR-M2 and VVR-M5 Fuel Assemblies-Operation Manual, 0001.04.00.000 РЭ, (2006).

[3] Report on the 7th Core Loading for the Dalat Nuclear Research Reactor, NRI, Dalat, (in Vietnamese), (2009).

[4] J.R. Deen, W.L. Woodruff, C.I. Costescu, and L. S. Leopando, "WIMS-ANL User Manual Rev. 5", ANL/RERTR/TM-99-07, Argonne National Laboratory, (February 2003).

[5] P. Olson, "A Users Guide for the REBUS-PC Code, Version 1.4," ANL/RERTR/TM02-32, (December 21, 2001).

[6] J. F. Briesmeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code, Version 4C", LA-13709-M (April 2000).

[7] John G. Stevens, "The REBUS-MCNP Linkage", Argonne National Laboratory, (2007).

[8] N.A. Hanan, J.R. Deen, J.E. Matos, "Analyses for Inserting Fresh LEU Fuel Assemblies Instead of Fresh Fuel Assemblies in the DNRR in Vietnam, 2004 International Meeting on RERTR, Vienna, (2004).

[9] V. V. Le, T. N. Huynh, B. V. Luong, V. L. Pham, J. R. Liaw, J. Matos, "Comparative Analyses for Loading LEU Instead of HEU Fuel Assemblies in the DNRR", RERTR Int'l Meeting, Boston, (2005).

[10] Teresa Kulikowska et al., Raport IAE-40/A, (1999).

[11] Arne P. Olson, M. Kalimullah, "A users guide to the PLTEMP/ANL V3.8 Code", ANL/RERTR, Argonne National Laboratory, (June, 2009).

[12] Le Vinh Vinh, Huynh Ton Nghiem and Nguyen Kien Cuong, "Preliminary results of full core conversion from HEU to LEU fuel of the Dalat Nuclear Research Reactor", RERTR Int'l Meeting, Beijing, (2009).

[13] Technical Design for Reconstruction and Enlargement of the Dalat Nuclear Research Reactor - Volume 3. State Design Institute, USSR State Committee for the Utilization of Atomic Energy, Moscow, (in Russian) (1979).

[14] Additional Physics and Thermal-Hydraulic Data for Reactor IVV-9, State Design Institute, USSR State Committee for the Utilization of Atomic Energy, Moscow, (in Russian) (1980).

[15] RELAP5/MOD3 Code Manual, SCIENTECH, Inc. Rockville, Maryland, (1999).

[16] W. L. Woodruff, et al., A comparison of the PARET/ANL and RELAP5/MOD3 Codes for the Analysis of IAEA Benchmark Transients and the SPERT Experiments, RERTR Program, ANL.

[17] V. V. Le and T. N. Huynh, Application of RELAP5/MOD3.2 for the DNRR, Proceedings of JAEA Conf. 2006-001.

[18] M. M. Shah, Improved General Correlation for Critical Heat Flux during Upflow in Uniformly Heated Vertical Tubes, International Journal of Heat and Fluid Flow, Vol. 8, No. 4, pp. 326-335 (1987).

[19] MACCS2 Computer Code Application Guidance for Documented Safety Analysis, U.S. Department of Energy, (June 2004).

[20] A.G. Croff, A User Manual for the ORIGEN2 Computer Code, Oak Ridge National Laboratory, (1980).

[21] INTERNATIONAL ATOMIC ENEGY AGENCY, Derivation of the Source Term and Analysis of Radiological Consequences for Research Reactor Accidents, SAFETY REPORTS SERIES No. 53, VIENNA (2008).

[22] Governmental Decree for the Implementation of the Ordinance on Radiation Protection and Control, Government of the Socialist Republic of Vietnam, No. 50/1998/ND-CP, Hanoi, (in Vietnamese) (1998).