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TRANSIENT/ACCIDENT ANALYSES FOR FULL CORE CONVERSION FROM HEU TO LEU FUEL OF THE DALAT NUCLEAR RESEARCH REACTOR

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ABSTRACT

Transient/accident analyses were performed in preparation for the conversion of the Dalat Nuclear Research Reactor from the use of Highly Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU) fuel. The analyses were done by staff at Dalat Nuclear Research Institute, Vietnam Atomic Energy Institute, with the support of the RERTR program at the Argonne National Laboratory. The initial core consisting of 92 LEU WWR-M2 (19.75% ^{235}U) fuel assemblies and 12 beryllium rods added around the original neutron trap was analyzed to evaluate its response to the transients/accidents caused by the initiating events of uncontrolled withdrawal of a control rod, cooling pump failure, earthquake and fuel cladding failure. Results of the analyses showed that safety of the reactor is maintained for all transients/accidents analyzed.

1. Introduction

The Dalat Nuclear Research Reactor (DNRR) is a 500-kW pool-type research reactor using light water as both moderator and coolant. It was reconstructed and upgraded from a 250 kW TRIGA MARK II reactor and put into operation in 1984 with the first fresh core consisting of 89 WWR-M2 fuel assemblies (FA) enriched to 36% [1].

In 2006 the DNRR was granted a permission to carry out the partial core conversion from the use of highly enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel, and on 12 September 2007 the first six WWR-M2 LEU fuel assemblies with enrichment of 19.75% were loaded in the reactor core [2]. Recently, the DNRR is safely operated with a mixed fuel core including 92 HEU FAs and 12 LEU FAs.

The feasibility study for full core conversion of the DNRR to LEU fuel was also performed by Nuclear Research Institute (VAEI) under the support of RERTR program at Argonne National Laboratory. The results of neutronics and steady-state thermal hydraulics calculations for full LEU core configurations, described in another paper at this meeting, showed that a LEU core loaded with 92 fuel assemblies and 12 beryllium rods around the neutron trap meets the safety requirements while maintaining the utilization possibility similar to that of the previous HEU and recent mixed fuel cores [3]. This paper presents results of the transient/accident analyses for the this LEU core.

2. Designed LEU Working Core

The geometries and materials of the HEU WWR-M2 fuel assemblies and LEU WWR-M2 fuel assemblies are described in Figure 1 and Table 1. Because the content of uranium in a LEU WWR-M2 fuel assembly is higher than that of HEU WWR-M2 fuel assembly, a LEU core with the same arrangement as the first HEU working core will not satisfy the shutdown margin requirement [4]. To overcome this problem, a 92 LEU WWR-M2 FA core added with 12 beryllium rod around the original neutron trap (Figure 2) was proposed and analyzed. The calculated results showed that this new LEU designed core meets the safety requirements from neutronics as well as thermal-hydraulics point of view. A summary of the core parameters used for the safety analysis is given in Table 2.

Figure 1. WWR-M2 Fuel Assembly

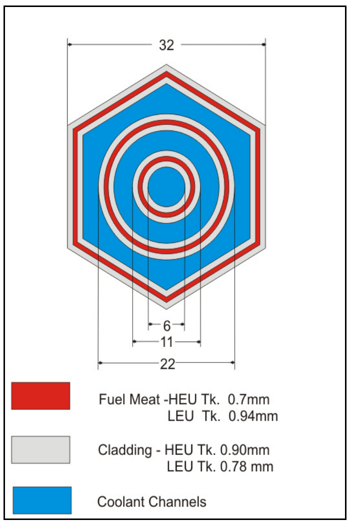

Table 1. Characteristics of WWR-M2 HEU Fuel Assembly and LEU Fuel Assembly

Fuel Assembly Parameter	VVR-M2 HEU	VVR-M2 LEU
Enrichment, %	36.0	19.75
Average mass of ^{235}U in FA, g	40.2	49.7
Fuel meat composition	U-Al Alloy	UO_2+Al
Uranium density of fuel meat, g/cm^3	1.4	2.5
Cladding material	SAV-1	SAV-1
Fuel element thickness, mm	2.5	2.5
Fuel meat thickness, mm	0.7	0.94
Fuel cladding thickness, mm	0.9	0.78

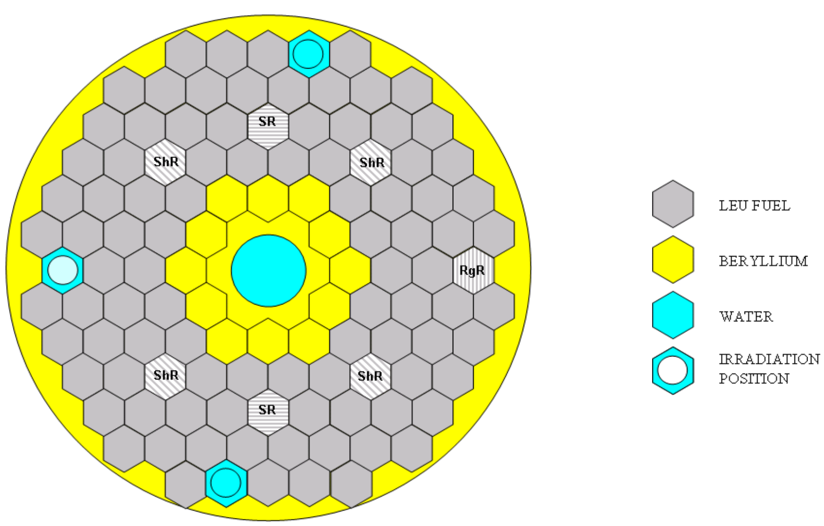

Figure 2. The new designed working core loaded with 92 LEU FAs

Table 2. Core parameters used for safety analysis

Parameters	Values
Power, kW	500
Coolant inlet temperature, ^{o}C	32
Peaking factor (shim rods at 300mm)	
- 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^{o}K)	-1.264×10^{-2}
- Fuel, %/^{o}C;	
(293-400^{o}K)	-1.86×10^{-3}
(400-500^{o}K)	-1.92×10^{-3}
(500-600^{o}K)	-1.56×10^{-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

3. Transients/Accidents Analyzed and Computer Codes Used

The following accidents scenarios, considered to be affected by core conversion, have been analyzed to evaluate the response of the reactor to the specified conditions:

(1) Uncontrolled withdrawal of a control rod,
(2) Cooling pumps failure,
(3) Earthquake, and
(4) Fuel cladding failure

Transient analyses are used for description of accidents (1), (2) and (3). The accident (4) in which the cladding of one fuel assembly is assumed to be stripped and the release of radioactivity occurs is defined as the Maximum Hypothetical Accident (MHA).

The transient analyses for the initial core consisting of 92 LEU WWR-M2 FA were performed using RELAP5/3.2 code [5]. The computer code is a generic 1-D thermal-hydraulics network code for simulation of nuclear and non-nuclear systems involving steam and water mixture, non-condensable and solutes, developed at Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission. The code validation of thermal-hydraulics and core dynamic characteristics of the DNRR was confirmed by comparing the analytical results with the experimental data [6].

In order to evaluate the source term and radiological consequences for the maximum hypothetical accident of the DNRR, the ORIGEN-2 [7] and MACCS2 [8] codes were used. The ORIGEN2 code was developed at Oak Ridge National Laboratory for calculating the buildup, decay, and processing of radioactive materials. It was used to determine the radionuclide inventories of the DNRR core. Because the original cross-section libraries of ORIGEN2 code are not suitable, the burnup dependent cross-sections of actinides and some fission products of the DNRR were calculated by MCNP5. The MACCS2 code is a Gaussian plume model for calculation of radiological atmospheric dispersion and consequences that could result from postulated accidental releases of radioactive materials to the atmosphere. The code was developed at Sandia National Laboratories (USA).

4 Results and Discussions

4.1. Uncontrolled withdrawal of a control rod

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

a) Start-up case:
   (1) -1% Δk/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.

4.1.1. Uncontrolled CR withdrawal at start-up condition

Reactor initially at 10^{-5}%FP and -1% Δk/k sub-critical

Figures 3 and 4 show the calculated results using RELAP5/3.2 code for the transient case when one shim rod is inadvertently withdrawn with speeds of 3.4 mm/s, from the core, which is initially sub-critical at 1%-reactivity depth and power of 10^{-5}%FP (5×10^{-8} MW). About 44.8 seconds after the initiation of the event, the reactor reaches to the fast period trip setting of 20 seconds, the scram signal on fast period appears and the reactor protection system functions after 9.1 seconds delay. The reactor power only increases to the maximum value of 2.78×10^{-7} MW while the fuel cladding temperature is unchanged.

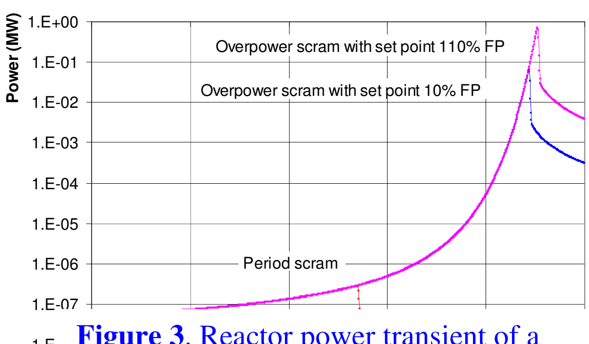

Figure 3. Reactor power transient of a shim rod withdrawal from 10^{-5}%FP and -1% sub-criticality

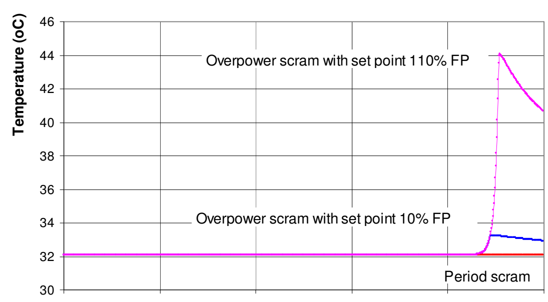

Figure 4. Cladding temperature transient of a shim rod withdrawal from 10^{-5}%FP and -1% sub-criticality

With the assumption of no fast period scram signal generated, the reactor is shutdown by overpower scram signal after 88.7 or 90.4 seconds since the initiation of the event with the overpower trip settings of 10% or 110%FP respectively. The reactor power reaches to the peak values of 6.21×10^{-2} MW (for overpower trip setting of 10%FP) and 7.18×10^{-1} MW (overpower trip setting of 110%FP) then sharply suppressed by the control rods insertion. The maximum fuel cladding temperatures for the above-mentioned cases are 33.3^{o}C and 44.1^{o}C respectively.

Reactor initially at 10^{-3}%FP and critical state

From a initial conditions of criticality with the power level of 10^{-3}%FP (5×10^{-6} MW), RELAP5/3.2 code was used to analyze the events of one shim rod or the regulating rod which are inadvertently withdrawn at the speeds of 3.4 mm and 20 mm respectively. The analytical results (see Figures 5-8) are not worse than the transient case of one shim rod withdrawal from -1% sub-criticality and power of 10^{-5}%FP except the event of the regulating rod withdrawal. In this case, 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 temperature (116 ^{o}C).

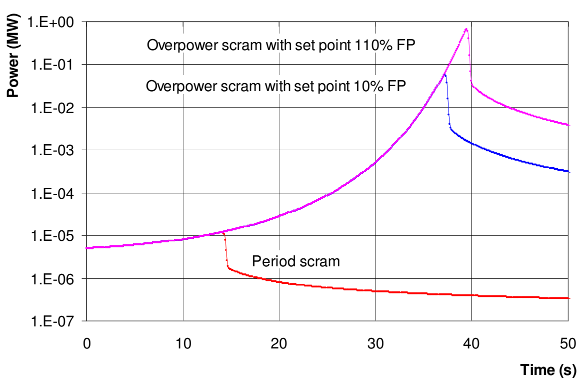

Figure 5. Reactor power transient of a shim rod withdrawal from 10^{-3}%FP, critical state

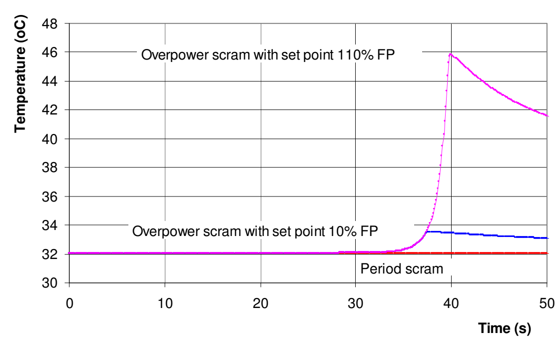

Figure 6. Cladding temperature transient of a shim rod withdrawal from 10^{-3}%FP, critical state

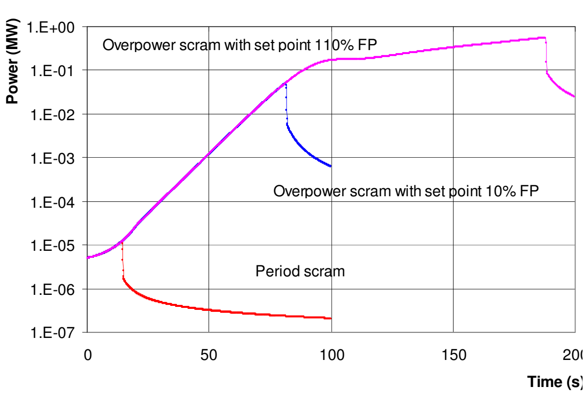

Figure 7. Reactor power transient of the regulating rod withdrawal from 10^{-3}%FP, critical state

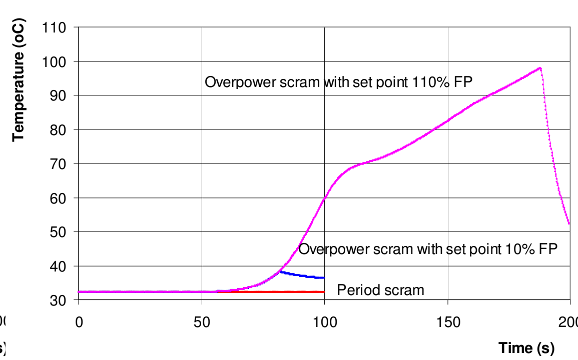

Figure 8. Cladding temperature transient of the regulating rod withdrawal from 10^{-3}%FP, critical state

4.1.2. Uncontrolled CR withdrawal at nominal power condition

RELAP5/3.2 code analytical results for 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 Figures 9 and 10. In this case, the reactor power increases and reaches to the over-power setting value of 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 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 appearance 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 seconds 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 estimated about 6.5 is much higher the acceptance criterion of 1.5.

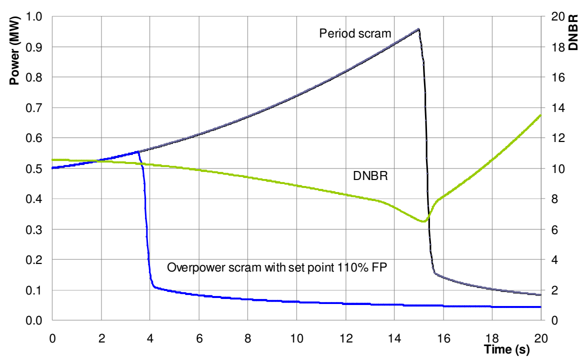

Figure 9. Reactor power transient of one shim rod withdrawal from a stable operation of 100%FP

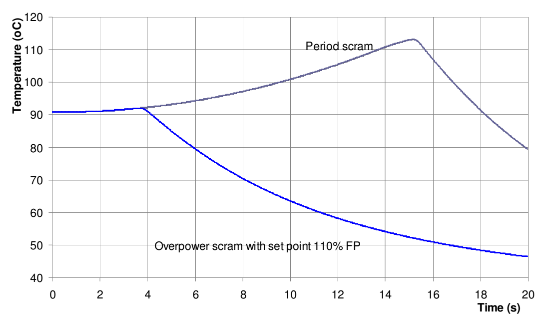

Figure 10. Cladding temperature transient of one shim rod withdrawal from stable operation of 100%FP

With the same initial conditions, the calculated results for the event of withdrawal of the regulating rod are showed in Figures 11 and 12 with very little differences comparing to 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.

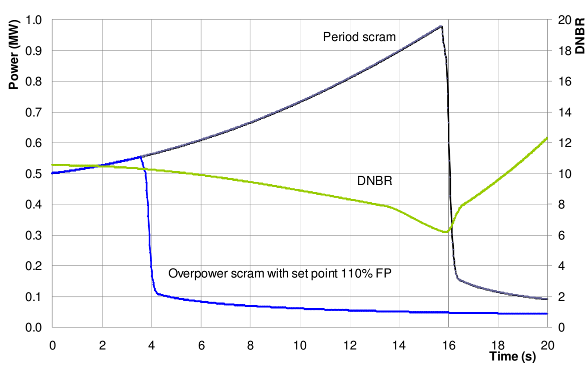

Figure 11. Reactor power transient of the regulating rod withdrawal from stable operation of 100%FP

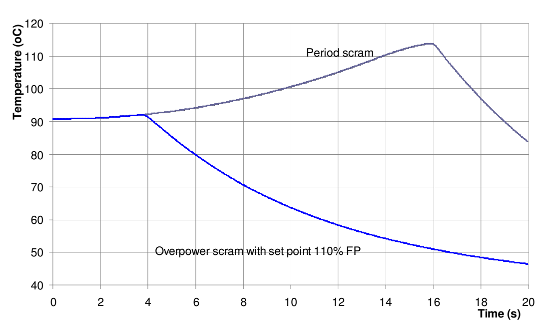

Figure 12. Cladding temperature transient of the regulating rod withdrawal from stable operation of 100%FP

4.2. 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 in Figure 13 show that the clad temperature reaches the maximum allowable operating clad temperature of 103 ^{o}C at about 55 min; i.e. the reactor could continue operation for 55 minutes within the envelope of the limiting conditions of operation.

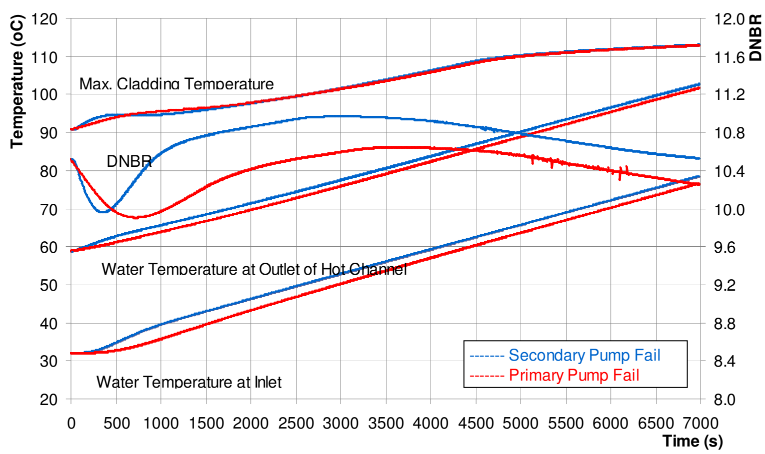

Figure 13. Transient of cladding and coolant temperatures in the cooling pumps failure event while the reactor power is remained operation at 100%FP

4.3. Earthquake

In case of earthquake, the most dangerous consequence for the reactor is that all control rods may be partially ejected out of the core, inserting a significant positive reactivity. For the DNRR site, the earthquake is estimated of intensity grade VI in the MSK scale.

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 and then decreases with the power, the residual heat from the fuel after raector shutdown is sufficiently removed by natural convection of pool water without considerable increase of the temperature.

Figures 14 and 15 show the analytical results of the earthquake event assuming the protection system fails to shutdown the reactor and the primary and secondary pumps stop operating due to loss of offsite power caused by the earthquake. 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 decreasing with no significant overheating of the fuel. The maximum outlet water reaches 89^{o}C and gradually decreases to a value of about 60^{o}C, which is still far below the saturation temperature. The min DNBR of 4.79 is much higher than acceptance value.

In case the cooling pumps remain working after the earthquake event (very unlikely), the analytical results are presented in Figures 16 and 17. 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^{o}C then gradually decreases to a stable value of 115^{o}C without nucleate boiling. The maximum temperature of outlet water is 89^{o}C at the peak power then decreases and stabilizes at about 82^{o}C, well below the saturation point. The min DNBR in this case estimated about 4.74 is still far from the acceptance criterion.

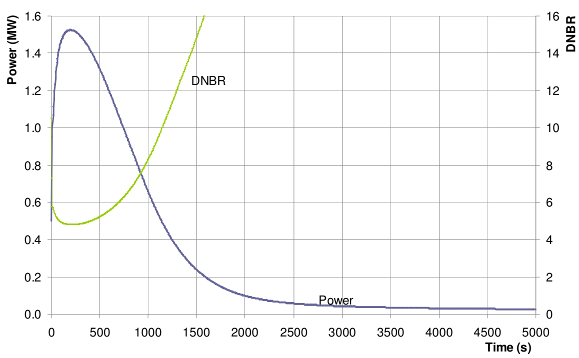

Figure 14. Power responses to earthquake event while cooling pumps are stopped functioning

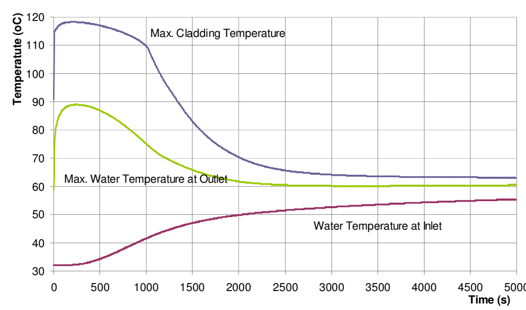

Figure 15. Temperature responses to earthquake event while cooling pumps are stopped functioning

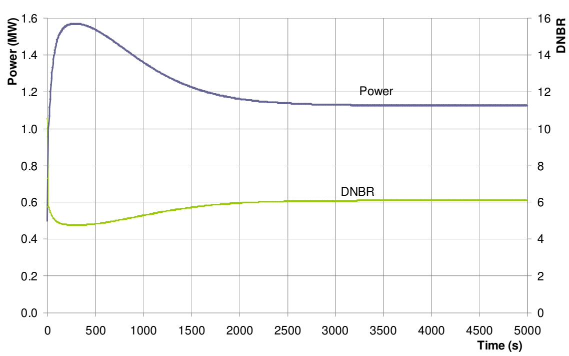

Figure 16. Power responses to earthquake event while cooling pumps are remained functioning

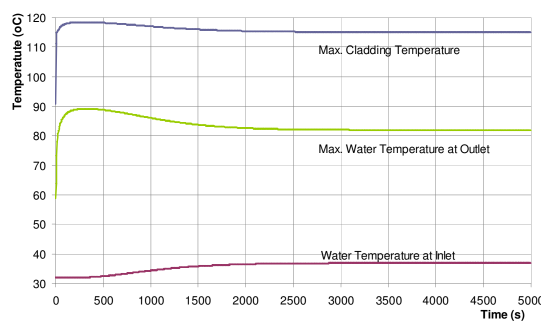

Figure 17. Temperature responses to earthquake event while cooling pumps are remained functioning

4.4. Fuel cladding failure

As stated above, an event of the strip of the cladding of one fuel assembly resulting in the release of fission products into the environment is postulated as MHA for the DNRR.

For the derivation of core inventory, it is 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 at full power with no decay. The fission product inventory of the damaged fuel assembly was calculated using the ORIGEN2 code.

From the calculated fission product inventory of the damaged fuel assembly, it is assumed that 100% of noble gases (Xe, Kr), 25% halogens (I), and 1% of other radionuclides (Cs, Te) [9] are released directly from the fuel to the reactor building with 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, using the following assumptions: (1) The radionuclides are released to the environment through the 40-meter high stack; (2) The Gaussian plume model is used to calculate air concentration of radioactivity; (3) Doses at each downwind distance are calculated for one year after the arrival of the plume; (4) 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 total effective dose equivalent, including cloudshine dose, inhalation dose and groundshine dose, as a function of the distance from the source is shown in Table 3. It can be seen that the total effective dose equipvalent to the public has the maximum value of of 0.64 mSv/year at distance about 450 m from the stack. This value is lower than the annual dose limit of 1.0 mSv specified for the public [10].

Table 3. The total effective dose equivalent vs distance for the MHA

Distance (m)	Calculated Dose (mSv/y)	Distance (m)	Calculated Dose (mSv/y)
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

5. Conclusions

The results of transient/accident analyses show that safety of the reactor is maintained for all transients analyzed. The result obtained for the Maximum Hypothetical Accident shows that the effective equivalent doses for the public is lower than the annual dose limit specified for the public.

These results of transient/accident analyses have been included in the Safety Analysis Report for the Full Core Conversion of the Dalat Nuclear Research Reactor to Low Enriched Uranium Fuel.

6. Acknowledgment

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7. References

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