VN9000001

NINR--E001-90

(IAEA-SM-310/78)

MAIN SAFETY LESSONS FROM 5-YEAR OPERATION OF THE RENOVATED DALAT NUCLEAR RESEARCH REACTOR

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Abstract

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The paper presents main safety-related characteristics of the Dalat Nuclear Research Reactor (DNRR), which was reconstructed in 1982 at the site of the former TRIGA Mark II, while retaining some of its structures. Experience acquired from reactor operation is analysed. The programme of investigations aimed at better ensuring nuclear safety of the reactor, together with some of its results are presented. Finally some propositions to improve the present situation are suggested.

I. Introduction

The Dalat Nuclear Research Reactor (DNRR) was reconstructed in 1982 while retaining some structures of the old TRIGA Mark II reactor. The latter was constructed in 1963 and operated until 1975. Just before April 1975, its fuel elements were removed from the reactor core and transferred elsewhere, making its operation definitely impossible.

The reconstruction of the new Dalat research reactor was carried out for 19 months, and adjustment works after its first criticality lasted more than 4 months. The reactor was inaugurated on March 20th, 1984 with a nominal power of 500 kW and since has been operating to realize the objectives attributed to it, namely training , research, isotope production and neutron activation analyses.

This article will first present main safety-related characteristics of the DNRR. Experience acquired from reactor operation will be analysed. Some results of investigations and studies aimed at better ensuring nuclear safety of the reactor will be presented. Finally, some propositions to improve the present situation will be suggested.

II. Main safety-related characteristics of the DNRR [1]

The Dalat reactor was redesigned and reconstructed at the site of the former TRIGA Mark II reactor, following general directions to secure economy in construction funds, adequate characteristics in nuclear safety and radiation protection, and good performance in some important physical parameters of the reactor (Cf. Table I), namely :

- Maintaining some engineered and technological structures of the previous reactor, such as concrete shielding walls, reactor tank, horizontal beam ports and graphite reflector.

- Keeping the natural convection mechanism for reactor core cooling.

- Doubling the reactor power and obtaining as high neutron fluxes as possible, thus allowing effective use of the reactor for radioisotope production and activation analyses.

Corresponding technical solutions consist of :

- The adoption of Soviet VVR-M fuel elements, enriched to 36% ^{235}U, and made of uranium - aluminium alloy. Each element is laid out in three concentric layers to increase the heat exchange superficies (Cf. Figure 1).

- The use of an additional reflector in beryllium to reinforce the neutron reflection capacity of the graphite reflector.

- The installation at the upper side of the reactor core of a cylindrical extracting well which has the effect of improving the natural circulation regime.

It may be worth mentioning the following safety related characteristics of the DNRR:

a) In order to ensure a large safety margin against the boiling of cooling water at the fuel element surface, the number of fuel elements has been finally fixed at 89 (Cf. Figure 2), while the reactor reached its first criticality with 69 fuel elements in the simplest configuration, and further with a central Beryllium-made neutron trap with 72 fuel elements. Hence it is clear that the DNRR has been operating with a significant value of excess reactivity (9$ at the beginning, and 6$ at present).

On the other hand, due to the enrichment as well as to the composition of nuclear fuel, although its Doppler coefficient of reactivity is negative, the DNRR does not possess as favorable prompt temperature coefficient as the former TRIGA reactor, which incorporated a zirconium hydride material mixed with fuel material in its fuel elements. Thus, in safety evaluation of the DNRR, it is no longer possible to totally eliminate the possibility of a reactivity accident as is the case with TRIGA reactors.

These are the two reasons which dictate the need for securing a very reliable control system for the DNRR. The system was completely redesigned and installed in order to meet such requirement. It comprises as many as 7 control rods symmetrically disposed inside the core, associated with 9 neutron detectors at the outside of the graphite neutron reflector, together with related electronic and mechanical systems.

b) To ensure radiation protection for all the staff working inside the reactor building, a layer of heavy concrete was added to the lateral shielding of the reactor at intermediate level to protect against gamma radiation from the reactor core which is now working with increased power, and also from strengthened upward water flow bearing activated elements produced in the reactor core. Also a steel cover was used for the reactor tank, in order to give better protection against upward radiation, for the same reasons.

The rather high level of radiation dose rate inside the old reactor tank had been unfavorable for the assembling process of the reactor core structures. Thus the solution of an entirely suspended structure was adopted for the reactor core.

The irradiation facilities in this core comprise 2 vertical pneumatic irradiation channels together with a central neutron trap and a wet irradiation channel. Surrounding the reactor core, a rotating tray containing 40 irradiation holes is arranged at the same position as the former "Lazy Suzan". Of the 4 horizontal neutron beam ports retained from the old reactor, one is used for experiences involving neutron reactions and also for neutron radiography, and a second is being arranged for neutron beam exploitation. An irradiation facility was also installed in the thermal column.

The introduction of a steel cover for the reactor tank, the suspended structure of the reactor core, the use of a reactor control system made of 7 control rods as well as the additional irradiation facilities inside the reactor core make the reactor's internal structure rather complicated in comparison with the former reactor. Handling operations in the reactor core become more subtle, and so do the operations aimed at inspecting periodically the reactor tank and other internal structures. These operations are even not practicable in some areas inside the reactor tank.

This is a problem of great concern, because structures kept from the old reactor are now more than 25 years old and their periodical inspection must be performed as rigorously as possible in order to detect any unacceptable defect if exists, and thus to prevent any incident leading to loss of integrity of the second barrier.

c) The conservation of reactor core cooling by natural convection is dictated by the desire to keep a cooling procedure with high degree of reliability, and also partly due to the fact that at 500 kW, forced convection is not yet absolutely necessary. Natural convection regime, as already stated, is improved by the installation of an extracting well. Anyway, it is necessary to be more concerned about the safety margin against water boiling at the surface of fuel elements, particularly in transition conditions, when power increases rapidly but the cooling water flow needs some delay to reach a new regime. Control of this safety margin in case of stable operation as well as in transition processes is absolutely necessary. Before results of such direct measurements are available, it has been felt prudent to ensure as large as possible a safety margin in case of stable operation at 500 kW.

III. Some experiences gained from reactor operation

a) After the physical start-up and initial adjustment of all technological systems, the reactor was put in operation and actually used for the realization of its attributed objectives. It can be affirmed that the DNRR has been safely operating from the beginning until now. The reactor never suffered any incident which really affected its safety.

Most unexpected scrams (Cf. Table II) were due to failures or incidents of the electrical network, and some others were due either to human errors or to failures of the reactor control system. The main reason of these latter failures is the strong influence of humidity on equipment installed in badly conditioned rooms, particularly in the rainy season.

It is obvious that equipment, system and components installed in such rooms invariably have poorer reliability in comparison with those benefiting good air conditioning.

It is worth noticing the poor quality of relays and motors such as pumps, fans belonging to reactor auxiliary systems. Due to climatic conditions, relays frequently suffer bad contact, affecting more or less the reliability of reactor control and technological systems. These faults can be easily repaired; however, maintenance and controlling measures must be reinforced, particularly for the equipment installed in unfavorable environment. Motors must also be well maintained and moreover their operating procedures must be rigorously followed in order to avoid hazardous consequences of their failure.

As the present reactor control system had been designed in the 70's, it did not benefit improvements brought later on. On the other hand, electric and electronic systems are not tropicalized to be capable of enduring harsh climatic condition of our region. The most significant incidents which happened to the control system were unwanted withdrawals of a safety rod. The first one occurred in the course of a maintenance operation (12/3/1987) and the second during system tests before a reactor run (23/9/1987). After the first incident, it was decided to stop the reactor for 6 weeks, during which the reactor control system was carefully examined but nothing was found which could explain the anomaly. After the second incident, it was thought that this could be due to a chance alien happening in the electronic circuitry, probably caused by short circuit. In October 1987, an electronic safe-lock was added to the motor driver of the safety rod AZ1 in order to hinder unwanted upward movement of the rod.

The incidents were considered as of safety concern but actually did not affect the safety of the reactor. After the technical measure adopted in October 1987, no similar incident has happened to the system since.

Failures in the electrical network were at the origin of a significant number of unexpected reactor scrams. Following the present design, when the reactor operates, an auxiliary power supply is continuously available from a diesel generator functioning at empty load and a second diesel generator is ready to replace the first should it fail. However this auxiliary system is only capable of supplying power for the reactor control system, lighting and some very restricted auxiliary systems. Power breakdown from electrical network inevitably leads to reactor shutdown and loss of cooling capacity via the primary circuit. But this is of no effect on the reactor safety.

Commutation to the auxiliary power system can be done with a good chance of success and allows to maintain the control capacity of the reactor.

b) From a management point of view, special concern has been devoted to the training of operating staff. All reactor shift heads and operators are nuclear physicists or nuclear engineers having received special training on reactor physics. Some of them have been trained abroad mainly in the Soviet Union for reactor operation, the others received their on-the-job training at the DNRI. Other shift members comprise a mechanician, an electronician, an electrician and a health physicist; they can be either engineers or technicians, but all have received special training both theoretical and practical. Every year all shift members have to pass examinations in special knowledge as well as in health conditions, in order to obtain the permission to work on.The knowledge controlled is related to fundamental and specialized matters, nuclear safety and radiation protection, and operational procedures in normal situation as well as in emergency condition.

A very important aspect in management is to make frequent inspections in order to ensure the observation of all safety regulations and operational procedures during shifts. Indeed, in reactor operation, everyone, aware of the importance of the activity, must impose self-discipline. This is constantly concerned with at any inspection, the result of which can have direct influence on one's immediate and future interest. The management experience from Dalat reactor operation clearly shows the efficiency of such inspection measure.

IV. Safety related research and investigation programme and some results obtained

From the very beginning, a research and investigation programme has been established and implemented in order to obtain more and more detailed knowledge of the physical characteristics of the reactor in normal as well as in emergency condition.

The programme of measurement is aimed at verifying the values of designed parameters, and periodically estimating the variation with time of physical parameters [2].

a) A measurement programme of the reactor physical parameters is being undertaken. The programme allows to acquire information about reactor core dimensions (such as effective height and radius), vertical and radial neutron flux distributions, differential and total control rod worths.

This programme also allows to investigate parameters which reflect the performance of the control system (in particular its protection function), such as the control rod drop time, and the time elapsing between the moment when a safety-related physical parameter of the reactor overpasses a definite limiting value, and the moment when the scram signal is emitted. Temperature coefficients of reactivity have been measured using the method of compensation with the automatic regulating rod previously calibrated.

The value of equilibrium Xenon poisoning effect has also been measured on the basis of controlling the vertical position of control rods at 60 hr after having reached the nominal rate.

The delayed photoneutron effect has also been investigated and evaluated. The Dalat reactor uses as reflector material an important quantity of Beryllium, so that delayed photoneutrons due to the reaction $^{9}Be(\gamma,n)$ lead to significant increase of the value of β_{eff}, hence affecting the dynamical behaviour of the reactor.

b) The research programme comprises the investigation of the reactor behaviour after promptly introducing determined reactivity values into the reactor core. After introducing 0.2 (resp. 0.4)$ reactivity into the reactor core with an initial power of 0.1 kW, the reactor power has reached equilibrium value of 100 (resp. 500) kW. As a result of this investigation, it is found unwise to introduce rapidly into the reactor core reactivity of values greater than 0.4$ (or equivalently 0.32%).[3]

This problem is also investigated on the basis of theoretical calculation. In particular, the hypothetical unwanted withdrawal of control rods has been studied by Soviet designers in cases of initial power of 500KW and of subcriticality (- 1% and at power of 10^{-7} of nominal value):

- a shim rod KS withdrawn at speeds of 3.4 and 1mm/sec.

- the automatic regulating rod AR withdrawn at speed of 20mm/sec.

Results of the calculation show that, if the protection function works correctly, the reactor remains safe in case of unwanted withdrawal of a control rod. However this result is uniquely based on power level protection ; should this protection fail, the power period protection would not be fast enough to ensure proper safety of the reactor core. Especially, in case of shim rod withdrawal at speed of 1mm/sec, the protection signal by power period would appear only after 30 sec. If the power level protection failed, at this delay, maximum temperature could reach 150°C and power could surpass 200% nominal value.

c) The event of primary circuit loss of flow, i.e. loss of heat evacuation capacity, has been studied by theoretical calculation. In the hypothetical case of primary circuit loss of flow without scram, if the reactor continued to function at nominal power, fuel surface temperature would reach 108°C (the boiling temperature of water at reactor core pressure) after 40 min. If the reactor power were maintained lower, the delay for reaching this temperature considered as dangerous would be higher.

V. Some propositions to improve the present situation

The safety status of the DNRR has been examined in detail when preparing its Safety Analysis Report (SAR). A step further towards a more quantitative approach is the application of Probability Safety Analysis (PSA) for safety evaluation of the reactor. This work is being performed and will be completed in 1990.

Safety analyses of the DNRR have led to the following propositions [4] :

a) It is of primary concern to ensure quality as well as reliability of the reactor control system. After more than 5 years of operation in unfavourable environment conditions and with more spare parts lacking, this control system begins to manifest some signs of obsolescence. The system needs to be renovated, some of its parts replaced using modern technology and tropicalized components.

Waiting for the realization of this renovation, it is reasonable to better ensure air conditioning in rooms where is located equipment of the reactor control system (as well as of some auxiliary systems).

b) The programme of general examination and maintenance of the reactor, and particularly an underwater telescope examination of the reactor tank and internal structures must be given favourable conditions to realize as completely as possible and in due time. Especially, spare parts must be urgently supplied in order to ensure quality of maintenance.

c) The fuel element surface temperature measurements will provide very important information in stable and also in power transition condition. A programme of such measurement by means of fuel elements equipped with adequate thermocouple is being implemented and results are expected to be soon available.

The realization of the above mentioned items is under the scope of DNRI activities in the safety field. Such activities need support of the responsible authorities, so that the DNRR can continue to be operated safely and reliably, realizing all its attributed objectives with higher efficiency.

The authors are indebted to Prof. Pham Duy Hien, Dr. Ngo Quang Huy, Dr. Vu Hai Long and Mr. Ta Tat Thang for their discussions and advice.

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Table I. Characteristics of Dalat Reactor

Parameters	Values/Nature
Power	500 kW
Fuel	U-Al alloy
Coolant and moderator	H_2O
Cooling	Natural convection
Number of fuel elements	89
Core loading	3576 g U-235
Enrichment	36 % U-235
Excess reactivity :	
- 1984, March	9 $
- 1989, July	6 $
Control rod worth :	
- Safety rods (2)	5.51 $
- Shim rods (4)	11.51 $
- Automatic regulating rod (1)	0.46 $
Rod insertion time	0.5 sec
Thermal neutron flux	
- Average	$4.x10^{12}$ n/cm^2.sec
- Maximum	$2.1x10^{13}$ n/cm^2.sec
Maximum surface temperature of fuel	98 °C
Maximum temperature of water in the core	52 °C
Effective delayed-neutron fraction	0.81 %
Temperature coefficient of reactivity	$-1.2x10^{-2}$ $/°C

Table II. Number of incidents during reactor operation causing reactor scrams

Incident type	1984	1985	1986	1987	1988
Electrical network deficiency	9	5	6	6	13
Equipment failure	4	4	1	1	1
Human error	4	1	4	2	3
Total	17	10	11	9	17

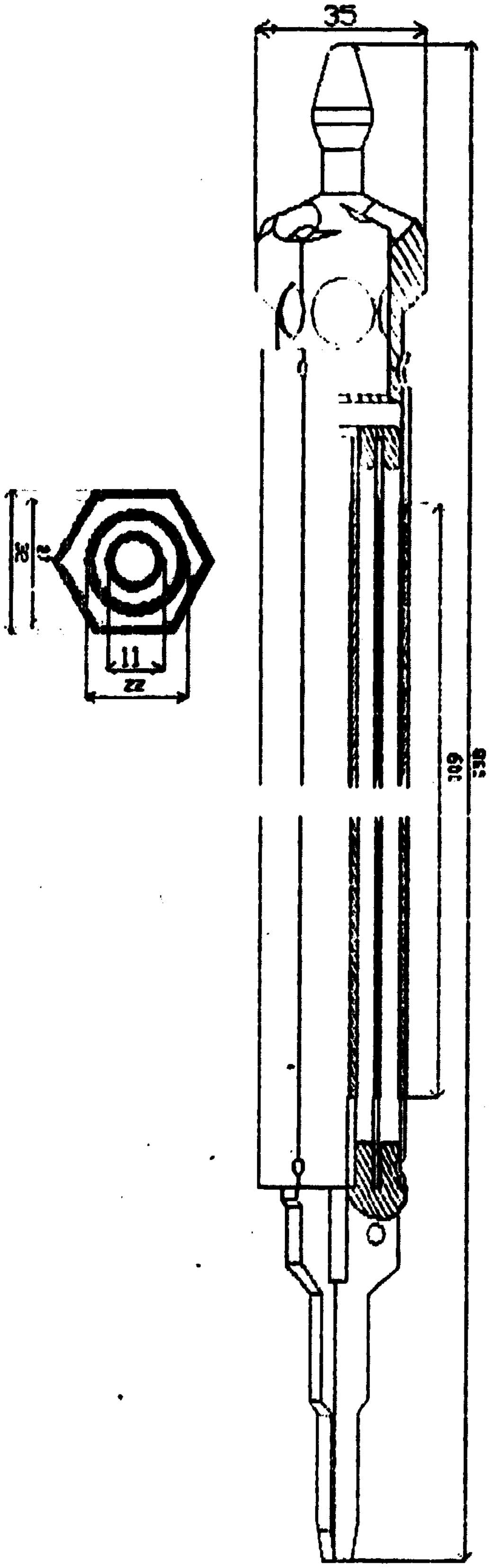

FIG.1 – FUEL ELEMENT BUNDLE

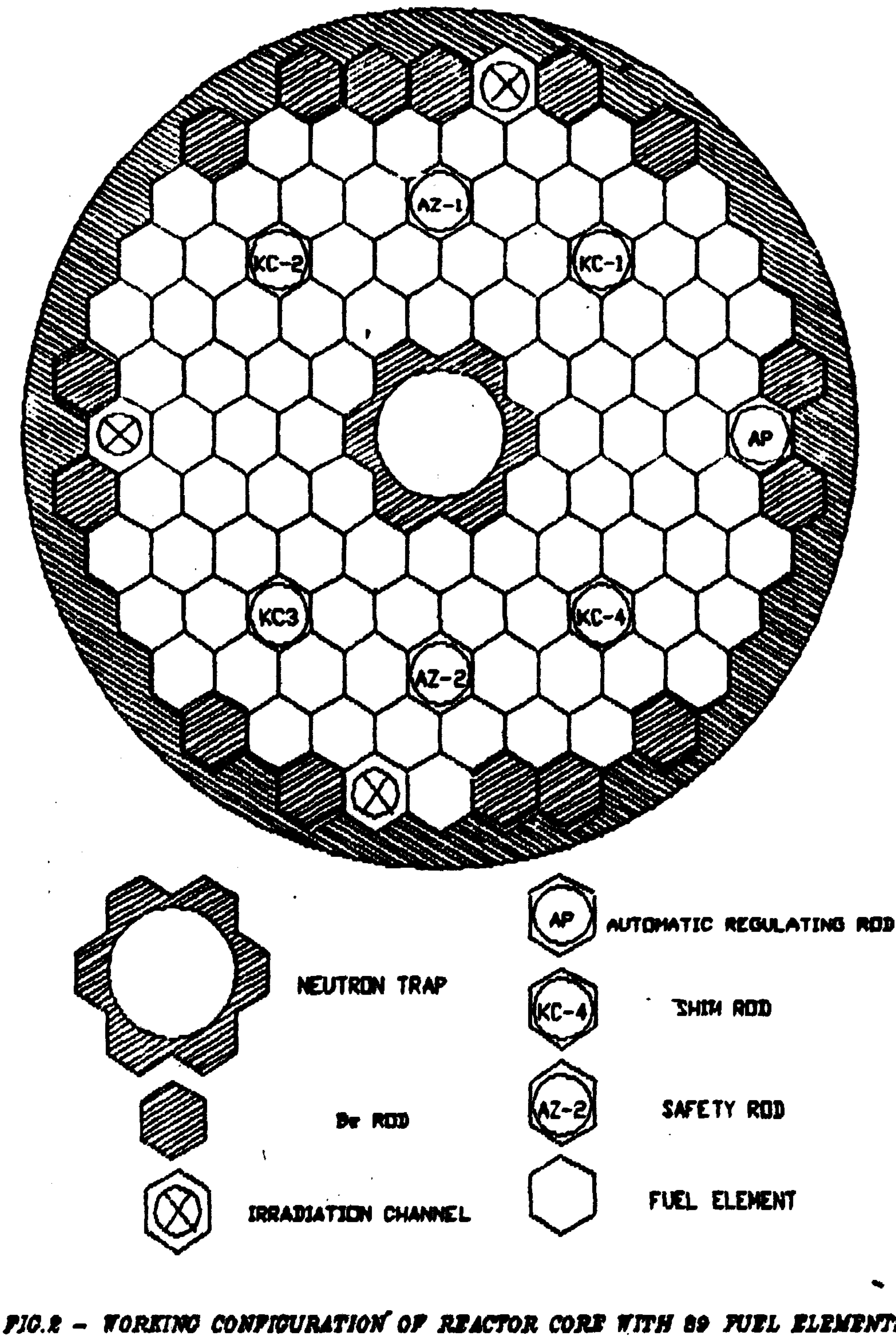

FIG.2 – WORKING CONFIGURATION OF REACTOR CORE WITH 89 FUEL ELEMENTS