Loss of off-site power accident analysis of Egyptian test and research reactor number 2

A. M. El-Messiry

National Center For Nuclear Safety and Radiation Control,
Atomic Energy Authority, Nasr city 11762, P.O.Box 7551, Cairo, Egypt

The chosen loss of Off-Site Power (LOSP) accident analysis of Egyptian Test and Research Reactor Number 2 (ETRR-2) covered the failure sequences of the Reactor Shutdown System (RSS), the redundant Emergency Diesel Generator (EDG) and the Uninterruptible Power Supply (UPS). Four generated LOSP accident scenarios are analyzed. Two codes are used in the calculations covered by the present paper. The first is the IAEA Probabilistic Safety Assessment Package (PSAPACK) used for constructing the relevant inductive Event Tree (ET) and quantifying the resulting sequence probabilities. The second code named TR22M21 has been developed by the author to simulate the specific nature of ETRR-2 under this accident. It determines the expected reactor behavior during the steady state and accident conditions. The two codes have been implemented in the four mentioned scenarios. It has been found that the two codes have successfully managed to identify a severe scenario of probability 0.01 that could significantly contribute to the risk of the core damage.

تستوجب مراجعة وتقييم أمان المفاعلات النووية إجراء تحليلات عندية لمجموعة من الحوادث ألتي يمكن أن يتعرض لها المفاعل تسمى الحوادث التصميمية . تحدث حادثة فقد الكهرباء بمصر بمعدل ١٠ مرات في السنة وهو معدل غير اعتيادي مما دفع لإجراء هذا البحث. توصى تعليمات الوكالة الدولية للطاقة الذرية فى الوقت الحاضر بإجراء تحليل حوادث المفاعل بكلا الطريقتين المعترف بهما وهما: الطريقة التحديدية(العندية) والطريقة الاحتمالية. حيث تحسب الطريقة الاحتمالية مدى احتمال حدوث الحادثة بينما تحدد الطريقة العندية قيم المتغيرات مثل قدرة المفاعل- درجة حرارته الضغط أثناء وقت حدوث الحادثة. طبق هذا المنظور البحثى على مفاعل مصر البحثى الثانى أثناء الحادثة المنكورة. تغطى الحادثة حالات تشغيل نظام أغلاق المفاعل ومحطة القوى الإحتياطية(الديزل) من عدمهما. درست أربع سيناريوهات مختلفة للحادثة. أستخدم فى الدراسة الإحتمالية كود عن الوكالة الدولية للطاقة الذرية وفى الدراسة التحديدية كود مصمم خصيصا من قبل المؤلف لتمثيل هذه الحادثة فى المفاعل المذكور. أوضحت الدراسة وجود سيناريو خاص من أحدى السيناريوهات المدروسة يؤثر تأثيرا مباشرا على خطورة أنصهار قلب المفاعل.

Keywords: Loss of off site power, Design base accident, Station blackout, Accident analysis, Nuclear safety

1-Introduction

A nuclear safety review and assessment of research reactors should include a deterministic analysis of the Design Base Accident (DBA) such as Loss of Off Site Power [1]. At present, it is recommend [2] to use both of the probabilistic method [3] to determine the likelihood of the accident and the deterministic method to establish and assertion the variation of the reactor safety related parameters, e.g. power, flow and temperature over the prescribed accident evolution time. This recommended approach has been implemented in the present work during LOSP accident to assess the safety of the Egyptian Test and Research Reactor Number 2 (new reactor). The LOSP frequency in Egypt is abnormally high; 10 times/year which give a good cause for initiating the present work. The description of accident scenarios and the related analysis are summarized.

2. ETRR-2 power system description

The basic structure of the electric power system of ETRR-2 reactor is shown in Fig. 1. It is classified as three power supplies, three loads and different switchboard [4].

2.1. Power supplies

2.1.1. Class C; normal power supply (NPS)

It consists of two independent external Lines L1 and L2 with medium voltage 11 kV, each is connected to an independent Transformer, T1 and T2, with output voltage

380/220 V and electrical capacity of 1500 kVA. T1 feeds the Low Tension Switchboard - /Left side/ (LTS-L) and T2 feeds the Low Tension Switchboard /Right side/ (LTS-R). Switchboards are cross-connected to each other. In normal conditions the NPS feeds Class C, B and A loads.

2.1.2. Class B; emergency diesel generator (EDG)

It is a small electric power station consists of two-diesel generators (DG) located at an independent area from the reactor building. They supply power to Class B and A for at least 24 hours continuous operation. Each DG has 200 kVA electrical capacity and 380/220 V output, it supplies power to either left or right class B. There is neither parallel operation of the DGs with the NPS nor between each others. The start up and connection between the DGs and class B busbars are done manually.

2.1.3. Class A; uninterruptible power supply (UPS)

It consists of a battery charger, batteries, DC/AC converter, and static by-pass switch. It is located inside the reactor building and has 10 kVA electrical capacity. It supplies power to class A loads for 30 minutes continuous operation.

2.2. Loads

2.2.1. Class C loads

Those loads which cause interruptions of the supply for indefinite time; e.g., normal lighting, control drive rooms, air conditioning system, water supply and treatment system, core cooling pumps and secondary cooling pumps.

2.2.2. Class B loads

Those loads whose reconnection to the system is a matter of convenience; e.g. ,emergency external lighting, ventilators, pool and auxiliary pool primary pumps, secondary circuit pumps for cooling pool and auxiliary pool water.

2.2.3. Class A loads

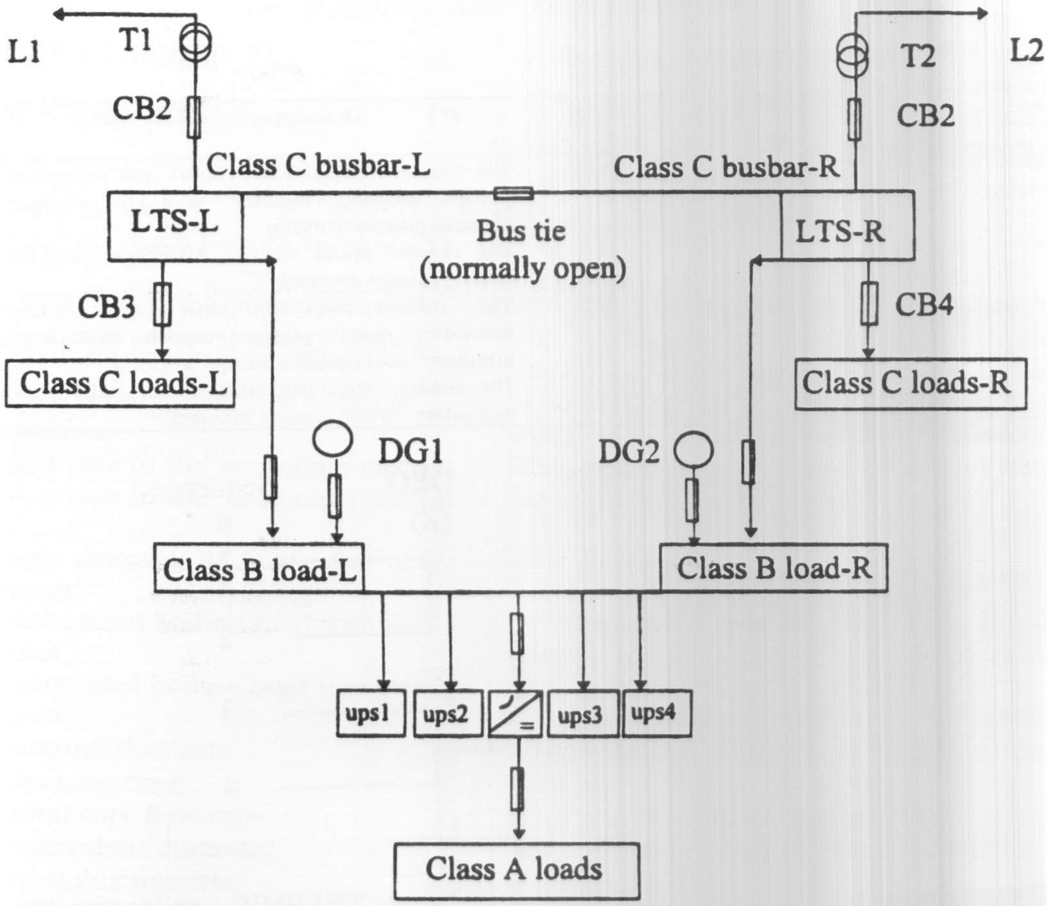
Those load which are essential from the safety point of view. They require secure power supply like UPS; e.g., instrumentation and control system, beacon lights, supervision and control system.

3. Accident scenario

As a result of LOSP accident no electricity is available to operate the entire primary and the secondary cooling circuits pumps (class C loads). Due to LOSP triggering signal, the Reactor Protection System (RPS) actuates the RSS that automatically insert the absorbing plates into the core to shut it down. In case of failure of the LOSP triggering signal the following derived triggering signals are activated [4]:

1. Low core pressure drop $\Delta P < 0.54$ bar,
2. High temperature difference across the core $\Delta T > 11$ °C,
3. Low core cooling flow $w < 500$ kg/s,
4. High core outlet temperature $T_o > 53.0$ °C,

Instantaneously the operator starts operating the EDG manually from the control room to provide an AC power for operating the reactor pool and auxiliary pool cooling pumps (class B loads). It also provides power to RSS (class A loads). Therefore the reactor pool will be cooled and a natural circulation loop will be established throughout the reactor core and reactor pool via an opening hole in the return pipeline of primary circuit. This hole is closed and opened by means of a flapper valve. The valve is closed by the upward force coming from the primary forced flow and opened upon loss of that force [5].



L1 &L 2: Utility source line 1& 2 — T1 & T2 : Power transformer 1 &2
CB1 &2: Circuit breaker 1 &2 — DG1&2 : Stand by Diesel generator 1 &2
UPS: Uninterruptible power supply — LTS-L & R: Low tension switchboard L & R

Fig. 1 ETRR-2 power system layout.

Four scenarios were generated for the studied case. Table 1 summarizes the state of the three power supplies for each scenario and the scenario description. In scenarios number 1 and 2 the reactor is shutting down due to LOSP signal or any of the derived signal and the reactor pool and auxiliary pool circuit pumps start working (depending on the availability of EDG). When the pool pump operates the pool and auxiliary pool cooling flow start increasing to its nominal values after approximately 5 minutes. During the accident both the primary and the secondary core cooling pumps are stopped, causing decay of the primary and the secondary flow. The decay rate depends on the pump coast down curve. In scenarios number 3 and 4 the reactor does not shut down due to failure of RPS as a result of UPS failure(class A supply) or other RSS failures.

4. Modeling

4.1. Probabilistic

Figure 2 shows the derived event tree model of the accident with the generated four accident scenarios. It describes the sequence logic diagram for each scenario. The initiating event (IE) is defined by the LOSP event in the event library, while the ET top headers are defined by RSS and EDG. The fault trees of the systems RSS and EDG are given in Ref. [4]

Table 1. LOSP generated scenarios.

#	Power supply C	Power supply B	Power supply A	Description of the scenario
1	N	Y	Y	The reactor shuts down, Primary and secondary circuit pumps stopped, Reactor pool and auxiliary pool circuits pumps working
2	N	N	Y	The reactor shuts down, All Primary and secondary circuit pumps stopped
3	N	Y	N	The reactor does not shut down, Primary and secondary circuit pumps stopped, Reactor pool and auxiliary pool circuits pumps working
4	N	N	N	The reactor does not shut down, All primary and secondary circuit pumps stopped

Y: Available; N: Not available

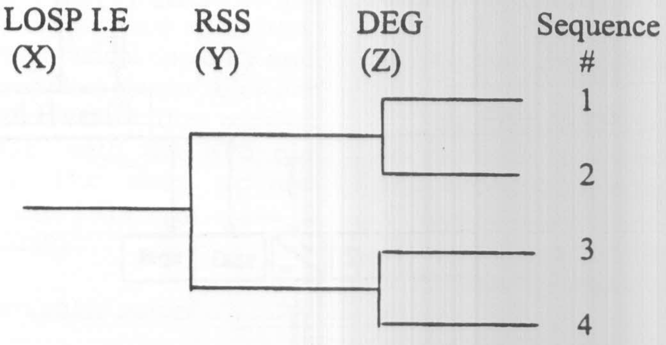


Fig. 2 LOSP event tree.

4.2. Deterministic

The computer code TR22M21 is developed to represent the specific nature of ETRR-2 during the steady state operation and LOSP transient conditions. The ETRR-2 model is shown in Fig. 3. The code simulates the core, primary forced cooling circuit, natural cooling circuit, and secondary cooling circuit. Two channels; hot channel (HC) and average channel (AC) represent the core. Every channel is divided into four sections [6]. The core power distribution is assumed cosine curve and averaged over each section. The pumps coast down curves supplied by the designer [5] are fitted by appropriate exponential formulas and introduced in the code. A natural circulation correlations [7] are incorporated. The physical part of the core is modeled by the point kinetic equation with feedback coefficients [8]. The resulting time-dependent physical and thermal-hydraulic equations are solved using finite difference technique in space. The code is verified against RETRAN and THERMIC codes for the steady state operation and showed good results [9]. The following are the main relevant equations:

$$W_p = W_{po} \exp\left(-\frac{t}{T_{tp}}\right), \tag{1}$$

$$W = W_n + W_p, \tag{2}$$

in which,

$$W_n = 0.0 \qquad t <= 70.0,$$

$$= W_{no} \frac{(t - t_n)}{t_{op}} \qquad 70.0 <= t <= 71.0 \tag{3}$$

$$= 9.722 \qquad t >= 71.0,$$

$$W_s = W_{so} \,.\, \exp\left(-\frac{t}{T_{ts}}\right), \tag{4}$$

$$h(i,j) = 0.023 \; \frac{K}{D_e} \; R_e^{0.8}(i,j) \; P_r^{0.4}(i,j). \qquad R_e > 2000 \tag{5}$$

A.m. El-Messiry/ LOSP analysis of ETRR-2

$$h_{bc}(i,j) = 0.074 \, [T_c(i,j) - T_{sat}(i,j)]^{2.68}, \quad (6)$$

$$h_{bs}(i,j) = U(i,j) + 2.253 \, [T_c(i,j) - T_{sat}(i,j)]^{2.96}, \quad (7)$$

Where:

- t is the time in sec,
- t_o is the time at which flapper valve starts open
- W_p is the flow rate in the primary circuit,
- W_{po} is the flow rate in the primary circuit at $t=0.0$,
- T_{fp} is the time constant of flow decay of primary circuit.
- W_s is the flow rate in the secondary circuit,
- W_{so} is the flow rate in the secondary circuit at $t=0.0$,
- T_{fs} is the time constant of flow decay of sec. circuit,
- h_{bc} is the subcooled boiling heat transfer coefficient,
- h_{bs} is the saturated boiling heat transfer coefficient,
- W_n is the natural flow rate
- t_{op} is the opening time
- W is the total core flow rate,
- D_e is the equivalent diameter,
- R_e is the reynolds number,
- P_r is the prandtel number,
- h is the forced heat transfer coefficient,
- T_{sat} is the saturation temperature,
- U is the coolant heat transfer coefficient,
- K is the thermal conductivity of water,
- $T_c(i,j)$ is the Wall temperature(clad)

Equation (6) applies for fully developed subcooled water at low pressure(2.07 to 6.3 bar), the units of $h_{bc}(i,j)$ and temperatures are Btu/(h ft^2 °F) and °F, respectively. Eq. (7) applies for fully developed saturated boiling water at low pressure (2 to 7 bar), the units of $h_{bs}(i,j)$ and temperatures are W/(m^2 °C) and °C, respectively.

5. Results and discussions

Each scenario is investigated using the two recommended methods i.e. probabilistic and deterministic.

5.1. Probabilistic

The RSS unavailability is taken as 10^{-3} [5], while the calculated EDG unavailability is 0.03. The resulting sequence numbers 1 to 4 are reduced and quantified using PSAPACK code with a cut-off probability of order 10^{-12}. The resulting sequence path and frequency or probability is listed in Table 2.

5.2. Deterministic

The main steady state data [6] for full power operation of ETRR-2 is given in Table 3.

The following Figs. 4 to 9 illustrate the response of ETRR-2 to LOSP transients, where:

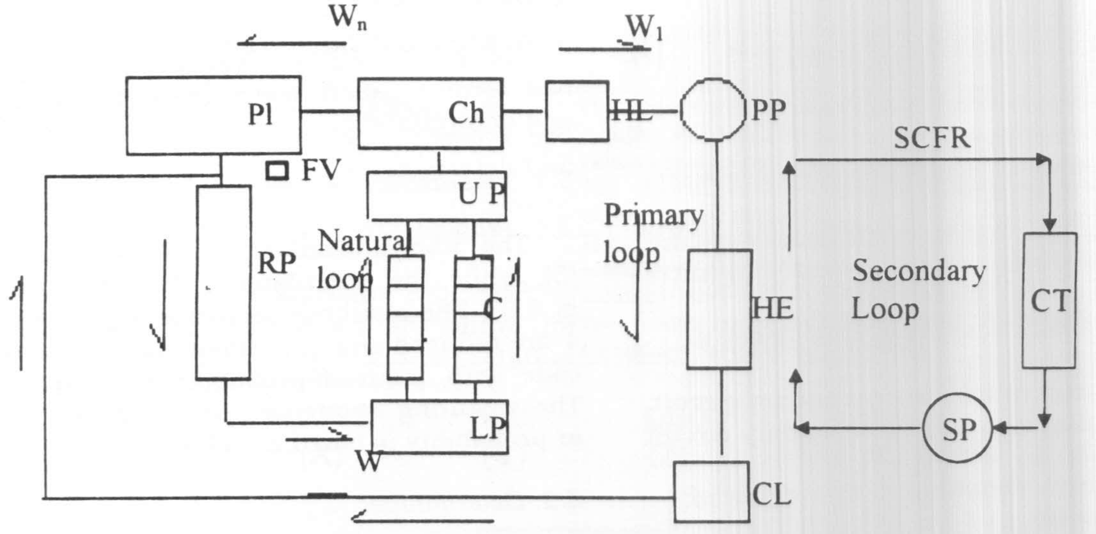
- P is the reactor power , MW,
- E_n is the energy released from the core, MJ,
- Tcd2 is the maximum clad temp- erature of the hot channel HC,
- Tcd1 is the maximum clad temperature of the average channel AC,
- Tc2o is the outlet coolant temperature of HC,
- Tc21 is the outlet coolant temperature of AC,
- Tsco is the outlet coolant temperature of the heat exchanger secondary side.

The discontinuity of the curves at t=4.0 is due to the large print out time step(4.0 s) compared with the calculation time step(0.0002).

Table 2. Event tree sequence result.

Sequence #	Path^a	Frequency(y^{-1})	Annual probability
1	Xyz	10	-----
2	XyZ	0.3	0.3
3	XYz	0.01	0.01
4	XYZ	0.0003	0.0003

a : Small letters means success, and capitals mean failure



C: Reactor core, UP: Upper Plenum, Ch: Reactor Chimney, HL: Hot Leg,
PP: Primary Pump, HE: Heat Exchanger, CT: Cooling Tower, CL: Cold Leg,
RP: Return Pipe Line, SP: Secondary Pump, LP: Lower Plenum, Pl: Reactor pool
FV: Flapper Valve

Fig. 3. ETRR-2 model.

Table 3. ETRR-2 full power operating data.

Parameter	Value	Parameter	Value
Power , MW	22.0	Average pool temp., °C	45.0
Core inlet temp. , °C	40.0	Cooling tower outlet temp., °C	30.0
Primary circuit flow W_1 , kg/s	527.78	Power peaking factor	3.0
Secondary circuit flow SCFR, kg/s	755	Natural circulation flow W_n , g/s	9.722
Primary flow decay constant, s	23.0	Secondary flow decay constant, s	32.0
Control rod worth, $	-11.86	Control rod insertion time , s	0.7
Mean pool temp., °C	45.0		

Figure 4. shows the decay of the primary loop flow W_1 and the secondary loop flow SCFR after LOSP accident. The shown curves are the fitted form of the coast down curves of the primary and the secondary coolant pumps supplied by the designer. The time constants for the decayed flows are stated in Table 3. The natural circulation flow W_n starts at time 68.9 s where the corresponding primary flow is 26.39 kg/s which is enough to open the flapper valve by its weight. At this instant the total core flow W begins increasing due establishing of the natural flow and then continue decreasing due to decrease of primary flow, and W approaches W_n at t=150 s.

Figures 5. and 6 represent the reactor response to LOSP transients for scenario #1. In the first few fraction of seconds before reactor scram the power peak of 22.2 MW occurs at 0.2 s while the Tcd2 peak of 96.15°C(which is below safety limit 105 °C) occurs at 1.4 s. The scram occurs at 1.4s when the flow decreases to 496.6 kg/s. As a result of reactor shutdown, the power drops to 0.69 MW at 4.0 s, while the energy release jumps to 84 MJ and then continue increasing with a small rate due to decay heat power after shutdown. Tcd2 drops rapidly and then increases to a maximum of 64.2 °C at 84 s, which demonstrates safe reactor shutdown. The Tsco decreases to a value 30.37 °C at 160 s due to decrease of reactor power and consequently heat transfer. Tc2o approaches 41.28 °C at 160 s. This scenario could happen with high annual occurrence frequency of 10 times/year, but no hazard could be expected and the core shuts down safely.

Figure 7 presents the reactor response to LOSP transients for scenario #2. The results are similar to the previous one with a little difference due to failure of EDG and consequently loss of pool water cooling circuit, therefore increase in its mean temperature by about 2 °C. This reflects on increasing the maximum Tcd2 to 46.6 °C at 84 s and Tc2o to 43.43 °C at 160 s. This scenario could happen with annual occurrence probability of 0.3, but no hazard may be expected.

Figures 8 and 9 show the reactor response to LOSP transients for scenario #3. The increase of temperatures due to failure of RSS is very high compared to that due to failure of EDG operation, hence results from scenario #3 and #4 are similar. The power decreases slowly due to negative temperature feedback reactivity coefficient resulting from rapid increase of the temperatures as a consequence of quick decrease in primary flow. The HC surface heat flux reaches 410.5 W/cm^2 at 47.24 s, which exceeds the burn out value 400 W/cm^2 . Therefore the clad and fuel could partially melt. At t=47.0 s, a little before the burn out point, the P, E_n , SCFR,W, Tcd2,Tc2o, go to values 6.13 MW, 653.8 MJ, 177.1 kg/s, 68.39 kg/s, 132 °C, 90.37 °C, respectively. Tsco increases to 36.93 °C at 16 s, then decreases slowly because of decreasing heat transfer power despite of decrease in secondary flow. This scenario which may happens with annual probability of 0.01 is considered very severe and contribute to risk from core melt.

6. Conclusions

The following concluding remarks are reached:

(1) Although LOSP event occurs frequently in Egypt 10 times/year, no hazard could be expected during operating of ETRR-2 such that the reactor shutdown system well function (scenario #1 with annual frequency of 10.0).

(2) Operation of emergency diesel generator following loss of normal power supply and reactor shutdown provides power to operate only the pool water cooling system and not the primary or the secondary cooling circuit (scenario #1). Hence, if the EDG fails the pool water temperature will increase by amount 2 °C. In such case no considerable increase of HC maximum clad and outlet coolant temperatures could be detected (scenario #2 with annual probability of 0.3).

(3) The severe LOSP accident scenario may occur with annual probability of 0.01(scenario #3) where the RSS fails. The result would be clad and fuel melts then core partial damaged, hence hazard to workers and peoples could be expected.

(4) Operation of EDG following loss of normal power supply and failure of reactor shutdown system does not affect the accident propagation or mitigate accident consequences.

References

[1] IAEA "Code on the Safety of Nuclear Research Reactors Design", Safety Standards SS-35-S1, Austria, Vienna (1992).

[2] IAEA "The Safety Assessment of Research Reactors and preparation of the Safety Analysis Report" , Safety Guide no 35-G1, Vienna; Austria (1993).

[3] US Nuclear Regulatory Commission "PRA Procedure Guide" , NUREG/C-2300, Vol. 1, January (1983).

[4] INVAP SEE "Safety Analysis Report of ETRR-2", Report # SAR 10, 0767-5325-3IBLI-001-10, NSRC, Cairo, Egypt (1997).

[5] INVAP SE "ETRR-2 Detail Engineering Design", Vol. LV, NC-NSRC, Atomic Energy Authority, Cairo , Egypt (1996).

[6] C. F. Obenchain "PARET A Program for the Analysis of Reactor Transients", IDO-172, US AEC (1969).

[7] J. P. Holman, Heat Transfer, Fifth Edition, McGraw Hill , New York (1984).

[8] Milton Ash, Nuclear Reactor Kinetics, McGraw Hill , New York (1980).

[9] A. M. EL-Messiry, ETRR-2 Control Rod Withdrawal Accident, under publication in Annals of Nuclear Energy Journal (1999).

Received December 25. 1999
Accepted March 6. 2000