FR 8603053

COMMISSARIAT A L'ENERGIE ATOMIQUE

INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE

DEPARTEMENT D'ANALYSE DE SURETE

CEA-DAS--216 R1

ON-SITE A.C. ELECTRIC POWER SOURCES FOR 900 MWE FRENCH NUCLEAR POWER REACTORS : RELIABILITY AND IMPORTANCE FOR SAFETY

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CSNI Specialist meeting on operation experience relating to on-site electric power sources

London (UK) 16-18 Oct 1985

CEA-CONF--8589

ON-SITE A.C. ELECTRIC POWER SOURCES FOR 900 MWe FRENCH NUCLEAR POWER REACTORS: RELIABILITY AND IMPORTANCE FOR SAFETY

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ABSTRACT:

After presenting briefly the new provisions laid down by the Electricité de France to meet a total electrical power loss, the main elements of the probabilistic study concerning the corresponding risk described: reliability data of internal sources used, results of risk improvement brought by the new measures, importance for internal source before and after implementation of the new measures.

RESUME :

Après avoir brièvement présenté les nouvelles dispositions prévues par Electricité de France pour répondre à une perte totale des alimentations électriques, seront fournis les principaux éléments de l'étude probabiliste du risque correspondante : données de fiabilité des sources internes utilisées, résultats de l'étude de risque, gain apporté par les nouvelles mesures, importance pour la sûreté des sources internes avant et après la mise en place des nouvelles mesures.

1. Introduction

When PWRs were introduced in France in the seventies, the approach adopted to safety was a deterministic approach based on the study of a limited number of conventional situations called sizing conditions for which the consequences of radioactive discharge had to be proved as being less than the limits set. The design of reactor safety systems was geared to observing these conditions and in addition, by applying the unique failure criterion.

Albeit this approach still lies at the base of French reactor design, since 1975, the necessity of including the examination of the loss of the main redundant safety systems, especially, reactor electrical power supply systems has made itself felt.

The Ministry of Industry specified in 1977 that power loss analysis should be based on a probablistic approach so as to situate the probability of unacceptable consequences with respect to the risk objective set at 10^{-7} per family of events and per year, and to propose any design changes that this may imply.

This approach is briefly described hereafter. After presenting briefly the new provisions laid down by the Electricité de France to meet a total electrical power loss, the main elements of the probabilistic study concerning the corresponding risk described: reliability data of internal sources used, results of risk improvement brought by the new measures, importance for internal source before and after implementation of the new measures.

2. Description of defences proposed by the Electricité de France

The EDF proposed a set of measures (Procedure H3) to cope with a total loss of electrical power. These measures cover the procedures and the means needed:

- to resupply the systems with electrical power in less than 3 hours via an ultimate source (for further information, see the paper by Mr. BERA),

- to repower the control and monitoring systems needed to handle the situation and to light the control room (installation of a turbo-alternator: for further information, see the paper by Mr. BERA),

- to automatically ensure primary pump seal cooling backup,

- to ensure a water supply to the primary circuit when it is open,

- to be able, under certain conditions, to reach a safe standby state that can be maintained.

Considering that an accident can occur during any one of the reactor states, the EDF has divided Procedure H3 into two parts:

- operating Rules H3.1: RHR not connected situations,

- operating Rules H3.2: RHR connected situations.

3. Reliability Data Used

Only the reliability data of the internal sources used in the French risk studies will be presented here.

3.1 Data Collection

Electricité de France has set up an organization dubbed "Système de Recueil de Données de Fiabilité (SRDF)" - Reliability Data Collection System - which will supply and calculate the reliability parameters concerning the components used in the French nuclear power station. This system was set up as soon as the six 900 MWe reactors in Fessenheim and Bugey were commissioned, for 1,500 equipment items approximately. The SRDF was gradually extended to the other power stations in operation. Currently, thirty one 900 MWe reactors and two 1,300 MWe reactors are connected to the system.

3.2 Diesel Power Generating Sets

The values given here are based on feedback and experience with the back-up power generating units in Fessenheim and Bugey from reactor commissioning up to early 1983. This 48 "year x diesel" cumulated experience represents 2,500 hours of operation for the generating sets for 1,350 start-ups and 400,000 hours on standby.

- Failure Rate in Operation

  Given the wide repair time dispersion, a mean value would be meaningless :

  λ_1: 1.2 10^{-3}/h (0.3 10^{-3} - 3 10^{-3}/h) $\tau_1 = 7$ h

  λ_2: 0.4 10^{-3}/h (0.02 10^{-3} - 2 10^{-3}/h) $\tau_2 = 6.85$ h

- Common Mode Failure Rate

  The SRDF can only be used to obtain an increase in the β factor. In fact, no failure can be taken into account as a failure with a common cause out of 27 complete failures.

  Hence: $\beta \leq \dfrac{0,7}{27} = 0.03$

- Start-Up Failure Rate

  γ = 1.7 10^{-2}/d (1.2 10^{-2} - 2.5 10^{-2}/d) $\tau_d = 10$ h

- Values Retained for Risk Studies

  .. In Operation Failure Rate Values from the SRDF are used as they are.

.. The Mean Time to Repair values are increased to take into account the difference between the repair times and the unavailability period.

Hence: $\tau_1 = 13$ h
$\tau_d = 13$ h

.. The Start-Up Failure Rate values are increased to take into account the possible unavailability of one of the diesel engine for preventive maintenance reasons.

According to the operating technical specifications, the unavailability due to preventive maintenance with the reactor operating is 46 h a year, hence:

$I_M = 5.2\ 10^{-3}$

It should be noted that only one of the two diesels can be subjected to preventive maintenance. Hence, the following start-up failure rate mean value has to be considered:

$$\overline{\gamma} = \sqrt{\gamma_1 \gamma_2}$$

where γ_1: Start-up failure of the 1st diesel generator either due to a fault or a programmed unavailability.

γ_2: Start-up failure of the second diesel generator due to a fault.

That is: $\gamma_1 = 2 + I_M$
$\gamma_1 = 1.7\ 10^{-2}/d$

Hence: $\overline{\gamma} = 2\ 10^{-2}/d$

These values are applicable for a 1 month interval between tests.

.. As to the failure rate with a common cause, the value $\beta = 0.1$, based on the American experience has been used so far.

The following value (start-up and operation) $\beta = 0.03$ has been retained in a recent survey by a CEA/EDF/FRAMATOME work group.

3.3. Gas Turbine Power Generating Set

In France, this type of equipment is new in nuclear power stations. As our experience is limited in this field, gas turbine power generating sets are put in the same category as diesel generators for the reliability data.

3.4. Operating on House Load

The value based on experience gives a success rate of 63%. This value is based on statistics for the French nuclear power stations over the years 1979, 1980 and 1981. In 1982, the success rate was 58% and 75% in 1983.

However, these figures must be considered carefully insofar as planned on house load operation has a success rate of 80%, while accidental on house load operation has only a 50% success rate. The last value is the value retained for risk studies.

Furthermore, the on house load operation failure rate in operation is arbitrarily (expert opinion) set at 10^{-3}/h. There is no justification for this rate insofar as successful on house load operation has always deliberately been interrupted after a short period (3.5 h maximum)

4. Serious Accident Risk Study

After reviewing the initiators considered in the risk probability study in the event of total power supply loss, the results of this study and the advantages of the new measures implemented will be assessed, and the equipment importance factor for the safety will then be defined with the evolution of this parameter before and after the introduction of the new measures.

4.1. Initiators considered

Given the backup means available in case of total loss of electrical power, this event is divided into two families. In effect, in the case of the loss of the 6.6 kV power supplies due to the loss of the redundant LHA and LHB switchboards, the ultimate electrical power sources (gas turbine or power from a Diesel set at another plant) cannot be envisaged as a defence. We have defined two families of events:

- loss of the two safety switchboards,

- loss of the sources.

Furthermore, if an accident occurs, the reactors may be in any one of the following states:

- run up, hot standby, hot shutdown ($T^{\circ} > 180^{\circ}C$ or $P > 45$ bars),

- shutdown on AFWS ($T^{\circ} < 180^{\circ}C$ and $P < 45$ bars),

- shutdown under RHR, SG available,

- shutdown under RHR primary open, water level at nozzle median plane,

- shutdown on RHR, open primary, pool full.

We have considered 5 states and 2 families of events, hence 10 initiators possible:

	RUN UP	SHUTDOWN ON AFWS	SHUTDOWN ON RHR	RCS OPEN LOW WATER LEVEL	RCS OPEN POOL FULL
Time in state/year	9 months	1 month	1 month	15 days	15 days
Sources loss	E1	E3	E5	E7	E9
Switchboards loss	E2	E4	E6	E8	E10

4.2. Results of the probability study on 900 MWe reactors

The calculation of the probability of the total loss of 6.6 kV power supplies was made using the fault tree technique. If the time spent in each of these states is known, the probability for each initiator can be determined.

INITIATOR PROBABILITY (PER YEAR AND PER REACTOR)	RUN UP	SHUTDOWN ON AFWS	SHUTDOWN ON RHR	RCS OPEN LOW WATER LEVEL	RCS OPEN POOL FULL
Sources loss	E1 1.65 10^{-5}	E3 3.6 10^{-6}	E5 3.6 10^{-6}	E7 1.75 10^{-6}	E9 1.75 10^{-6}
Switch-boards loss	E3 3.1 10^{-6}	E4 2.9 10^{-6}	E6 6 10^{-7}	E8 3 10^{-7}	E10 3 10^{-7}

Evaluation of the risk of backup electrical power being lost was made using the event tree method.

The probability of the sequences leading to core meltdown is given in the following table for each initiator and the reactor states considered.

PROBABILITY OF MELTDOWN (PER YEAR AND PER REACTOR)	RUN UP	SHUTDOWN ON AFWS	SHUTDOWN ON RHR	RCS OPEN LOW WATER LEVEL	RCS OPEN POOL FULL
Sources loss	E1 5.0 10^{-8}	E3 0.8 10^{-8}	E5 1.2 10^{-8}	E7 1.1 10^{-8}	E9 ε
Switch-boards loss	E8 3.3 10^{-8}	E4 ε	E6 5.4 10^{-9}	E8 7.9 10^{-9}	E10 ε
	8.3 10^{-8}	0.8 10^{-8}	1.7 10^{-8}	1.9 10^{-8}	ε

Or an overall risk of 1.3 10^{-7} year x reactor (error factor 10).

4.3. Gain from H3

To calculate the gain in risk limitation using H3, it is necessary to calculate the probability of serious accident without Procedure H3 and its associated systems.

The value of the corresponding risk is equal to 1.1 x 10^{-5}/year x reactor.

The overall gain provided by Procedure H3 can be estimated at:

$$G_{H3} = \frac{1.1\ 10^{-5}}{1.3\ 10^{-7}} \simeq \boxed{85}$$

4.4. Importance of safety systems

To evaluate the importance of various safety systems, we propose reasoning on the basis of the increase in risk due to the unavailability of a system. Importance can be characterized by the ratio of the annual risk increase due to the unavailability of a system over the risk objective per family of events (10^{-7}).

Importance factor

$$I = \frac{\text{Increase of risk due to unavailability of a system}}{\text{Risk objective per family of events}}$$

The importance factors are given for the on-site electric power sources. Two cases are distinguished: without and with H3.

	WITHOUT "H3"		WITH "H3"	
UNAVAILABLE SYSTEM	INCREASE IN ANNUAL RISK	I FACTOR	INCREASE IN ANNUAL RISK	I FACTOR
Diesel generator	1 10^{-4}	1000	1.5 10^{-6}	15
Ultimate sources (gas turbine, diesel generator of another plant)	-	-	4 10^{-7}	4
Turbine alternator ("LLS")	-	-	10^{-5}	100

The following can be established:

1) The importance of a diesel generating set for the safety has significantly decreased since additional facilities have been installed to counteract the effects of a power supply loss.

2) The importance of the diesel generating sets and backup sources is almost the same.

3) The addition of the LLS turbine alternator to counteract in particular the immediate effects of power supply loss is more important for the safety than new additional power sources which become in fact redundant after 3 hours (time required to put them into operation). The turbine alternator has a double function:

   . Power supply the standby injection pump for the seals

   . Power supply the command control required for condition follow-up and reactor fine control.

It should be stressed that criminal acts are not taken into account in the probability studies. Taking into account in a deterministic way the loss of external power supplies through criminal acts may lead to giving more importance to the standby power supplies: diesel and gas turbine power generating sets.

5. Conclusions

The probability studies made by the Institut de Protection et de Sûreté Nucléaire have shown that the value for the probability of core meltdown resulting from total loss of electrical power is around 10^{-7}/reactor/year for 900 MWe PWRs, with an error factor of less than 10. This result led the safety authorities to decide that the measures proposed by the EDF to counter electrical power loss were acceptable.

These studies also enabled quantification of the importance systems to safety based on the determination of the importance factor for each system.

Concerning operating rules, the safety authorities position is to take the importance factor into account for specifying scheduled testing.

Given the values of this factor, it has been decided that the LLS turbine alternator and the gas turbine generating set should be subjected to periodical tests.

This importance factor is also used for determining the permissible operating time in the event of system unavailability. In practice, these times are calculated with a risk increase tolerance of 10^{-7} per unvailability case. Under these conditions, their values are the reverse of the importance factor for the safety.

Finally, studies are to be made to determine the programmed unavailabilty rule (due especially to maintenance operations during reactor shutdowns for reloading) to hold the reactor induced risk at an acceptable level.

The various results given show the advantage of probabilistic risk analyses for the designer and user as much as for the safety expert: the quantified results provide a strong basis to direct the design choice, set the operating rules depending on the importance of the equipment, and assess the safety of the installation.

Avril 1986

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