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Assessment of the feasibility of an improvement programme enabling operation of units 3 and 4 of Kozloduy nuclear power plant

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Abstract: .

Since ten years, different western assessments have been made of the safety of VVER 440/230 units, including those of the KOZLODUY Nuclear Power Plant located in Bulgaria. Concerning the latter, reference can be made to the work of an European Consortium (GRS/Germany, IPSN/France, AEA/England, AVN/ Belgium), in 1992 and 1993, during examination of restart conditions for Units 1 and 2. The objective of these assessments was the improvement of the overall level of safety with a view to maintain these units in service with satisfactory safety level for a limited period. The work identified their main deficiencies.

At the present time, all the modifications decided upon for Units 1 and 2 of the KOZLODUY Nuclear Power Plant in 1992 and 1993 have been adopted and implemented for Units 3 and 4.

In 1999, considering that these units would probably not be definitively shutdown before about ten years, IPSN decided to perform an internal assessment of the feasibility of an improvement programme enabling continued operation of units 3 and 4 of KOZLODUY Nuclear Power Plant during this delay.

1 INTRODUCTION

The assessment of the existing VVER/440/230 reactors has shown that the current safety level is not sufficient. Thus, large improvements are necessary to ensure an acceptable safety level. The submitted report describes the assessment of Kozloduy 3 and 4 Nuclear Power Plants by identifying the main lacks in their design. The scope of requirements is established, keeping in mind that these NPP might not be stopped in the next few years. A dedicated assessment of the feasibility of an improvement programme has been undertaken by IPSN in this scope. This assessment is performed on the basis of Kozloduy 3 and 4 but can be extended to other plants of VVER/440/230. However, this document makes no allowance for any financial considerations associated with the accomplishment of an improvement programme.

2 SAFETY GOALS

The safety level achieved by the units after implementation of a contingent modernisation programme should be close to the internationally recognized INSAG 3 principles. Consequently, a large amount of studies have to be performed using generic and recognised approaches. At the end, using the results of these studies, the objectives of the article 25 of the INSAG 3 should be met: "The target for existing nuclear power plants consistent with the technical safety objective is a likelihood of occurrence of severe core damage that is below about 10-4 events per plant operating year. Severe accident management and mitigation measures should reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response."

These objectives constitute a target to define the main requirements of the safety approach to be fulfilled to define a convenient modernisation programme.

3 SCOPE OF THE ASSESSMENT

3.1 Extension of reference accident

In order to verify the sufficiency of the systems to fulfil the safety functions it was necessary to establish a new list of accidents considered to be probable and then to integrate them in the Design basis accidents list.

The first requirement is to cope with the largest diameter primary circuit pipe (500 mm). It is however reasonable to accept that conservative margins are required only for the breaks up to 200 mm which are representing the largest diameter pipe excluding the primary loops. Thus, no excessive conservatism is required for the assumptions to be used for the LOCA 500 mm study. It is, nevertheless, necessary to verify that core damage is avoided for this accident and that the confinement integrity is maintained as well. In addition, a low likelihood of the 500 mm break has to be substantiated by the demonstration of high quality of the piping, the extensive in-service control and the leak detection.

Furthermore, when updating the list of design basis accidents, particular care needs to be paid for:
- dilution accidents,
- accidents occurring during the different phases of outages,
- accidents which can lead to bypassing of the containment.

It is obvious that the major efforts have to be done regarding the core melt prevention. Nevertheless, different types of scenarios with regard to the containment integrity and eventual deterioration of tightness have to be studied and eliminated as far as possible.

They concern:
- 1st category : core meltdown with bypassing of the containment,
- 2nd category : core meltdown at high pressure (risk of Direct Containment Heating),
- 3rd category : core meltdown with hydrogen explosion inside the containment,
- 4th category : core meltdown with slow rise of pressure and temperature in the containment.

The allowance for accident scenarios involving core meltdown requires a pragmatic approach combined with the probabilities (PSA1&2) of the scenarios. A judgement will be made accounting the associated uncertainties in the PSA results and in the potential radiological consequences of the scenarios.

It is important to recall the objective of INSAG 3: "The target for existing nuclear power plants consistent with the technical safety objective is a likelihood of occurrence of severe core damage that is below about 10-4 events per plant operating year. Severe accident management and mitigation measures should reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response."

The impact on the overall risk of the measures which would be proposed in a modernisation program must be assessed. To do so, it would be helpful to obtain a Level 1 probabilistic quantification classifying the sequences leading to core meltdown according to the Level 2 consequences which could be envisaged (at low pressure, at high pressure and with bypassing of the containment).

Consequently, using the results of the probabilistic study, it is necessary:
- firstly, using the scope of potential sequences to demonstrate the sufficient elimination of release in the short term,
- secondly, to re-examine the compatibility of severe accidents with possible measures taken in the context of an off-site emergency plan.

The scenarios involving release in the short term (1st and 2nd categories of scenarios) need to be examined in depth to assess the sufficiency of the measures to prevent them (design measures as well as accident management measures).

To be able to judge the sufficiency of the preventive measures, it would be useful to examine all the corresponding sequences with an individual core melt probability higher than 10^{-7} per plant operating year. This value allows in particular to take margins in the examination of the corresponding very important sequences.

For the 4th category of sequences, studies have to be performed to define the needed mitigation measures. Obviously, the current leak rate (approximately 300% per day at a pressure of 1.3 bar absolute, i.e. approximately 1800 Nm^3/h) is too high. It is absolutely necessary to reduce the leakage of the containment to a reasonable value, and it must be possible to recover the remainder by dynamic containment systems (ventilation and filtration).
Moreover, for all the types of scenarios leading to deflagration or detonation of hydrogen, studies need to be carried out and adequate measures need to be proposed when necessary.

3.2 Reactor pressurised vessel

The possible service life of Units 3 and 4 of Kozloduy Nuclear Power Plant depends mainly on embrittlement of the pressure vessels, which must be the subject of meticulous overall assessment. It is necessary to confirm the trends observed in the results of tests on specimens and to ascertain that no major problems are revealed concerning the pressure vessels of Units 3 and 4 of Kozloduy Nuclear Power Plant. A detailed situation report on the pressure vessels must be submitted. The pressurised thermal shocks must be considered in the analysis.

3.3 Confinement (tightness and integrity)

It is important to reduce the existing leaks from the hermetic zones of Units 3 and 4 of Kozloduy Nuclear Power Plant. It requires to examine the compatibility between severe accidents and the measures taken in the context of an off-site emergency plan drawn up for Kozloduy Nuclear Power Plant.

The following functions have to be fulfilled:

- to maintain dynamic containment in the rooms adjoining the hermetic compartment,
- to filter out a sufficient amount of the radioactive elements present in the gases that leak out.

It is also necessary to maintain the integrity of the containment and to limit the value of the maximum pressure peak corresponding to the integrity maximum pressure for the entire range of primary and secondary breaks considered in the Design Basis Accident list.

3.4 Protection against internal and external hazards

The following has been assessed:
- fires,
- internal flooding,
- effects of high-energy pipe breaks,
- internal missiles, particularly in the event of disintegration of a turbine generator, and falling objects,
- earthquake,
- severe climatic conditions (extreme cold and extreme heat),
- aircraft crashes,
- flooding by the Danube River,
- external explosions and the risks resulting from the industrial environment.

For the internal hazards, the approach is to find the common points in the existing layout of the safety related items. An appropriate separation or if needed, adequate solutions are to be proposed.

3.5 Large «primary to secondary» leaks (collector break)

Experience has shown that the design of the VVER-440/230 steam generators is vulnerable. The problem is relative to the tops of the primary headers and of the manifold. Their rupture would result in a leak of water from the primary to the secondary side, with an estimated equivalent diameter of 100 mm. For some types of VVER 440/213 reactors, solutions have been proposed to reduce the leak cross-section to make it possible to manage such an accident. The application of these solutions on VVER-440/230 NPP has to be examined.

3.6 Qualification and ageing

For the equipment whose qualification is not substantiated, there are two possible approaches:
- establishing and carrying out a programme of qualification,
- replacement by qualified equipment.

Furthermore, additional equipment which might be installed to cope with Beyond Design Basis Accidents must be the subject of requirements established on a case-by-case basis.

A dedicated study should be done in order to establish an anticipating replacement of obsolete equipment (I&C, cables, ...) with regards to its life duration.

3.7 Safety Analysis Report

It is necessary to establish a comprehensive safety analysis and to formalise it in the SAR.

4 CONCLUSION

The NPP has not presented a modernisation programme. IPSN experts consider that, if such a programme would be achieved, the main topics presented above should be highlighted. In particular, in such a case, some items would need in depth investigations. They are related to:

- the present status of the vessels specifically for unit 3,
- the residual life time of important equipment such as I&C,
- the means to ensure the containment integrity under maximal pressure for the envelop case of DBA or BDBA,
- the effects of the uncertainties on calculated radioactive releases with respect to the corresponding objectives,
- the classification of the 200 mm primary break and the set of assumptions to be applied to study this accident,
- the value of leaktighness of the containment reasonably achievable taking into consideration the filtered leaks partition,
- the preventive measures taken to fulfil the objectives of the INSAG 3,
- the approach to be followed for severe accidents,
- the level of priority of the studies concerning treatment of internal and external hazards.