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REQUALIFICATION OF THE STEAM SUPPLY SYSTEMS OF UNITS 3 AND 4 OF THE KOZLODUY NPP TO A NEW MODEL WWER-440/B-209M

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Abstract

In order to achieve significant advance in operational safety level, the project characteristics, the possibility of safety systems upgrading and operational conditions of Units 1 to 4 of the Kozloduy NPP were an object of very serious and in-depth analysis in the years 1990-2000. This systematic evaluation was initiated under the broad international concern resulted from the conclusions of IAEA missions held during 1990-1991 to assess the safety of the units. As a result of the efforts of the plant staff and many international experts the operational conditions, design safety and plant management were dramatically improved which resulted in bringing the plant to a new safety level. This review also developed such that the design safety features of Units 3 and 4 are significantly different from those units of the so-called V-230 group. The principle difference and advantages of Units 3 and 4 design were clarified and confirmed. A review process of the changed status of Units 3 and 4 safety was conducted in 1999-2000 with the help of IAEA experts and the experts of RISKAUDIT and WENRA. The process led to the conclusion that the significance of advantages of the safety level need to be encapsulated within a new safety case and the corresponding set of steps was combined as a "Project for upgrading the Nuclear Steam Supply System of Units 3 and 4 of Kozloduy NPP to model WWER-440/B-209M". The completion of the activities under this project is expected in 2002 following the major implementation phase during 2001/2002 units' outages.

1. THE BASIC IDEAS OF PROJECT PR-B-209M

Earlier review of the design safety features of Units 3 and 4 confirmed that they are significantly different from those units of the so-called V-230 group. In contrast to the units of B-230 type, a three trains structure of safety systems is used. The capacity of emergency core cooling systems is defined by the condition to avoid significant fuel rods damage in case of main pipeline rupture in primary circuit. Many other technical solutions are implemented in the project in order to achieve higher safety level. These are the emergency control room, separated control systems for the active safety systems etc.

These good design solutions were significantly developed and expanded in sequentially implemented programs – "Short term safety improvement program" (1991-1997), "Complex program for units' upgrading" (since 1998 up to 2000), "Program for preparation and conducting of OSART mission in 1999". The implementation of these programs created the base for essential changes in the views for safety level of these units.

The aim of the new project PR-B-209M is to implement the necessary technical measures, analyses and design documentation changes for units' design basis change and to demonstrate the correspondence of new project to the contemporary safety requirements for the necessary internationally accepted level. The project consists of set of technical measures to address the remaining safety issues with regard to the solutions already internationally accepted and a special design review study for comparison and justification of the changed design status, taking into account contemporary design safety requirements.

The project was launched in 2000 on the basis and as a continuation of the Complex Program for modernization of the units thus assuring the consistency of the approach for continuous upgrading of the units safety.

The provision of core cooling with all initiating events that are related with primary circuit rupture is included in the safety analysis review of new project B-209M. The demonstration of not exceeding the limits for radiation impact on the environment in case of such events is also included. The acceptance criteria and conditions for the correspondent analysis are reasoned in each particular case in correspondence with the recommendations of IAEA and WENRA experts.

Guillotine brake of primary pipeline with diameter 209 mm is accepted as a base for demonstration of the capacity of core cooling systems, their single failure tolerance with conservative initial conditions and for system modelling. This is in correspondence with the approach accepted by IAEA.

The specific aims regarding the decisive aspects of unit safety are presented in next paragraphs:

- Reliable fuel cooling (in order not to reach conditions for fuel rod damage) has to be demonstrated for all postulated initiating events, related to brake of primary circuit pipelines with diameter 200 mm. This suppose using of deterministic approach (conservative initial conditions, considering single failure, acceptance criteria as for design base accident), including the radiological consequences. The accepted initial and limiting conditions have to be reasoned in the analyses. The conditions for preserving the fuel rod integrity are defined in the applicable documents as follows:
  - temperature of fuel cladding — not higher than 1200°C;
  - local depth of fuel cladding oxidation — not more than 18% of initial wall thickness;
  - percentage of reacted zirconium — not more than 1% of its quantity in the core.
- The initiating event with main circulation pipeline break (diameter 500 mm) is analysed using realistic approach regarding the initial conditions and availability of normal operation systems and safety systems. It has to be demonstrated for this initiating event that non-permissible change of fuel rod geometry in the core (melting, plastic deformation) is not reached.
- The problem with providing of fuel cooling for initiating events with LOCA is related directly with the requirement for guaranteed integrity of the last barrier – the construction of containment, and with satisfying the defined criteria for acceptable radiation impact of the accidents that are presented below.
- An approach using supplementary emergency feedwater system is implemented for Units 3 and 4 regarding the initiating events without loss of coolant. As a result the program measures lead to minimizing the probability of core damage. The basic sequences defining this probability are determined using the methods of probabilistic safety analysis and minimizing their impact is in the base of providing a general core damage frequency less than 10^{-4}.
- Preserving of concrete hermetic compartment integrity has to be demonstrated for all initiating events and with dynamic loads during the accident. A filtered pressure relief system will be implemented for pressure limitation. This solution is equivalent with the solution for pressure limitation used in B-213 project (bubble condenser).
- The leak of steam generator compartments has to be limited gradually by the project to not more than 100% volume/24 hours for the conditions of tightness test that is performed each year. At the same time, the maximum permissible untightness of the compartment is defined using the given below limits for radiation impact in case of accident.
- Satisfying of criterion for beyond design accident in accordance with BNSA requirements has to be demonstrated for all initiating events related to a rupture of real primary circuit pipeline with DN 200 mm. The criterion for initiating event with a rupture of pipeline DN 500 mm is not exceeding the limits for radiological impact for beyond design accidents.
- The accepted limits for the maximum forecast individual dose of population are defined as follows:
  — For design base accidents - effective dose of 50 mSv during the first year after the accident and absorbed dose of 150 mGy for the thyroid of a person on the border of preventive measures area;

— For beyond the design base accidents - effective dose of 5 mSv during the first year after the accident and absorbed dose of 50 mGy for the thyroid gland of a person on the border and out of the area for protective measures of 'Kozloduy' NPP.

- It is foreseen an implementation of a set of measures for filter modernization of the design ventilation systems in order to minimize the radiation impact on the environment during normal operation.
- The approach of the updated revision of program PRG'97A is used. It includes focusing on activities in the preventive measures' area - analysis of the real condition in order to define the strength margin, identifying the potential mechanisms for degradation and initiating of corrective measures if necessary. A number of strength and safety analyses are planed for that purpose. The necessary measures for providing the main pipelines reliability during seismic event are implemented too.
- A progress of the measures for early leak detection is planned using three qualified systems based on different principles. At time being one qualified and three not qualified systems on different principles are used. A progress in metal control as an organization, technology etc. is planned too.
- The approach of the updated revision of program PRG'97A is used. It includes continuation of risk analysis and completion of the necessary measures for equipment qualification, seismic strengthening of the constructions considering the updated seismic assessment, additional measures for fire safety etc.
- An implementation of a number of measures and development of a complete strategy for severe accidents' management are included in the project. These are reactor level measurement, forced filter ventilation of the compartment, installation of hydrogen recombination system etc.
- An upgrading of the design base of project B-230 units to project B-209M is achieved as a result of the different measures' implementation and satisfying the specific aims. In order to prove the bringing of original design base to the improved level, a technical document for implementation of the pack of project measures has to be issued for each unit separately.

2. SCOPE OF THE PROJECT

In general, 62 technical measures are included in the project. They are distributed in the accepted 13 groups' system considering the measures' impact on the aspects of safety provision in NPP. They are developed on the base of NPP operational experience considering the specific requirements for satisfying the aims that are defined above.

The most of measures (35) will be implemented by the end of 2001. Another 25 measures have to be completed in 2002. The last three measures, partial implementation of which is provided during units' outage, will be completed in 2003.

The total costs of project implementation are evaluated as above $66 000 000 and the funding is provided completely by NPP investment program. All opportunities for additional foreign funding (using PHARE program, for instance) shall be used on occasion.

3. THE PROCESS OF REQUALIFICATION

Based on the above presented approach in the beginning of the year 2001 the formal process of transferring KNPP units 3 and 4 to a new model was initiated jointly by KNPP and leading Russian design and scientific organization including the Main Designer. The process itself comprises of three stages:

- Development of a methodology for evaluation of the new units safety case and for review of the upgraded plant from the point of view of existence of effective protective leers. Agreement on this methodology by Safety Authorities;
- Performance of the evaluation according the agreed methodology and preparation of justification for the acceptance of the new safety case and consequently, of the new model. Presentation of this justification to the Safety Authority;

- Preparation of specification of the set of design documents representing the new model and introducing the changes into plant documentation.

A technical specification and work plan to perform the necessary steps were approved together with a schedule that foresees all work for justification to be done till the end of 2001.

4. CONCLUSION

The successful finalization of the project "Project for upgrading the Nuclear Steam Supply System of Units 3 and 4 of Kozloduy NPP to model WWER-440/B-209M", together with the already implemented modernization of Bohunice NPP Units 1 and 2 shall be considered as a great success of the concentrated efforts of the international community, IAEA and the corresponding countries toward improvement of safety in nuclear installations built on earlier standards and shall be used as a guiding example for other eastern countries.