TOP SAFE '98
Valencia - Spain, April 15-17, 1998

RESULTS OF NOVOVORONEZH-3 PSA PERFORMED WITHIN TACIS-91 PROGRAMME

J A Carretero (Empresarios Agrupados)
Y Shvyryaev (Atomenergoproekt)
G Tokmachev (Atomenergoproekt)
V Morozov (Atomenergoproekt)
P Fernбndez (Empresarios Agrupados)

ABSTRACT

This paper is a brief overview of the Level 1 PSA Study for Novovoronezh Unit 3 carried out within the framework of the European Commission Tacis programme. Both the objectives and the scope of the study are presented. The paper discusses the methodology used to perform the PSA for internal initiating events and for internal fires. The sources of the input data are defined, and the results in terms of core damage frequency are provided. Major contributors to core damage frequency are identified. Some insights and recommendations are also presented.

FULL PAPER

BACKGROUND

A Level 1 Probabilistic Safety Analysis (PSA) for Novovoronezh Unit 3 referred to as VVER-440 model 179 (in the family of 440/230 reactors) was carried out within the framework of the European Commission Tacis programme. The project was started in 1993 and was completed in 1997.

The main Russian counterparts for the project were Atomenergoproject (AEP) as the Russian leader with the collaboration of Gydropress, the Kurchatov Institute, and Novovoronezh NPP. The study was supported by a Western consortium led by Empresarios Agrupados from Spain with the cooperation of Belgatom from Belgium, and AEA Technology and NNC from the UK.

The objectives of the project were:

- Transfer of Western PSA technology to Russian organisations
- Estimation of the core damage frequency
- Identification of initiating events and selection of the dominant accident sequences having the highest contribution to core damage frequency
- Identification of possible operational safety improvements

The scope of work consisted of a Level 1 PSA, including:

- Internal initiating events during full power operation
- Internal fires during power operation

A deterministic pilot study was also performed on initiating events such as internal flooding, missiles, pipe whip, jet impingement and external events, which was limited by methodology transfer purposes.

METHODOLOGY USED AND DATA

The Novovoronezh PSA model was developed using a small event tree - large fault tree linking methodology, based on the Risk Spectrum PSA Version 2.10 computer code capability. The base model was developed to reflect the plant design status at April 1, 1994.

The definition of initiating events was based on plant specific data, VVER experience, generic sources, and other PSAs. In addition, an FMEA of support systems was performed to find potential initiating events specific to the Novovoronezh design.

Initiating events were grouped into 4 LOCA groups and 10 transient groups. Beyond design basis initiating events were not in the scope of the study.

To define the success criteria, all the thermohydraulic analyses available for this reactor type were used. No new thermohydraulic analyses were performed to verify PSA assumptions for the development of the event tree. They were confirmed by reference to a design basis analysis or by further calculations carried out by Gydropress and the Kurchatov Institute.

Initiating event frequencies were derived from the following sources:

- LOCAs: World-wide operational experience [1]
- Interfacing LOCAs: Specific analysis using simplified models
- Transients: Plant-specific operational experience. A total of 17 reactor-years of operating experience for Novovoronezh

VVER-440 was available from accident event reports. In some cases, VVER-440 reactor-type operational data was used as prior information in the Bayesian updating process [2]

- Common cause initiators: Fault tree analysis

Novovoronezh NPP plant-specific data was used to estimate the reliability parameters for the majority of components as well as for the unavailability due to testing and maintenance. A total of 13.5 reactor-operating years of experience were available from plant records. For cases where no plant-specific data was available, generic data were used from IAEA database for VVER [3] and from US generic databases [4].

Dependent failure analysis involved elaborate modelling of functional dependencies, subtle failure analysis, and common cause failure modelling. However, the possible fact that components and piping would not be able to withstand secondary side breaks was not considered. VVER-specific common cause failure data was not available for the study. Therefore, the Beta-factor method was adopted and the values used are generic.

A detailed analysis of human actions was carried out as part of the PSA study. This analysis involved pre- and post-accident human interactions. The THERP procedure was used for the quantification of pre-accident human actions. Modelling of post-accident operator actions was based on HCR and Swain's models.

Recovery action modelling was performed for the following actions:

- Recovery of power supply following LOOP
- Recovery of normal heat removal
- Closing of BRU-V relief valves and SG safety valves following their opening

The emphasis of the internal fire study was on the analysis of the consequences of a fire given that an initiating event could be induced, combined with the possible loss of safety-related systems. The internal event risk model was modified to account for the degradation of equipment as a result of a fire.

Twenty-six critical fire areas were identified for a detailed analysis of the possible consequences of a fire caused by a fire-induced initiating event and a failure in the system of interest.

The estimate of frequencies of each plant fire zone started by the identification of specific types of fire initiators such as cables, electrical wires, oil, and motor-operated mechanical equipment for which fire frequencies were derived from generic data. The frequency of a fire for a given zone was estimated as the sum of the fire frequencies from specific types of fire initiators multiplied by their weighting factor. The weighting factor of each initiator was determined by the relative fraction of fire initiators located in the specific fire zone.

A specific approach was used to evaluate fire consequences in selected zones. This included a screening analysis of fire scenarios based on considerations such as frequency and consequences.

The modelling of the zone fire scenarios was based on the assumption that a fire would damage all cables and equipment in the location except for pipes, heat exchangers and manual valves which are not supposed to be susceptible to fire. Different scenarios concerned with failure modes such as failure-on-demand and spurious actuation were considered.

The fire consequences were classified into the following three categories:

- Direct core damage
- Internal initiating events with partial loss of safety functions
- Internal initiating events without any failure in the safety systems

Special attention was paid to fire zones assigned to the first category. For such fires, a detailed analysis was performed using the following non-conservative engineering judgement:

- For large fire areas, fires were assumed to be confined within the fire zones separated from each other by a distance of 6 metres, although special attention was paid to the spill of large amounts of oil from the turbine oil tanks

- For a smaller distance, a conditional probability of fire propagation was derived

In general, the possibility of fire propagation and suppression was analysed in a simplified manner. No computerised model was used to calculate the propagation of a fire. The possibility of a fire spreading from zone to zone was not taken into consideration.

RESULTS AND RECOMMENDATIONS

The results of the PSA for the current plant status is 1.82×10^{-3} per year for core damage frequency for internal full power initiating events, without consideration of beyond design basis initiating events.

The most significant contributors to core damage frequency in terms of internal events were found to be loss of off-site power, leakages of steam lines and loss of normal heat removal on secondary side. Figure 1 shows the distribution of major contributors for Novovoronezh Unit 3.

Internal fires also represent a very significant contribution. The overall fire-induced core damage frequency is estimated to be 3.84×10^{-3} per year. Among the fire scenarios, the main contributor to core damage frequency (about 90%) are fires that lead directly to core damage due to beyond design basis initiating events and/or loss of system functions. Of these a major contributor is a fire in the turbine hall, at about 62%.

Several points can be derived from the analysis of these PSA results.

1. More detailed thermohydraulic analyses have to be performed in order to reduce limitations of PSA performed.

2. There is no information about environmental qualification of the equipment in the event of a break in the steam header/collector or steam pipes outside confinement. This can be a very important factor in the case of steam isolation valves and the steam semi-collectors' segregation valves. A qualification programme should be established to understand how this equipment functions.

3. Detailed documentation/information about I&C systems is required in order to assess its risk contribution.

4. The fire-induced core damage frequency is mainly determined by poor separation between systems in the turbine hall. Therefore, arrangement of separated fire zones in the turbine hall was strongly recommended.

5. For internal events, PSA results are dominated by a certain number of design weaknesses identified in the cooldown system, secondary pressure control, emergency power supply, and the steam generators' feedwater isolation valves. The risk to the plant can be significantly reduced by the upgrading measures recommended by PSAs. These recommendations include:

   · Development procedures for periodical testing of cooldown system valves. Modification of testing procedures has already been implemented in Novovoronezh NPP

   · Provision of an additional long-term decay heat removal path. An additional technological condenser needs to be implemented to make the cooldown system single failure proof

- Provision of additional reliable means for supplying feedwater to the steam generators. A supplementary independent emergency feedwater system needs to be implemented which should have its own dedicated feedwater source

- Provision of heat removal capability by primary bleed-and-feed. A complete set of engineering analyses are required to provide this capability

- Provision of reliable isolation of charging lines to reduce interfacing LOCA risks

- Provision of motor-operated valves to automatically isolate an affected steam generator from the auxiliary feedwater system. Modification of the feedwater isolation actuation system is also required

- Provision of isolating capability in case of BRUV valve closure failure in the secondary side

- Provision of an automatic connection from diesel generators to essential buses in case two diesel generators fail to start

- Provision of additional capacity of emergency batteries to extend heat removal capacity in case of station blackout

- Provision to improve reliability of isolation of confinement ventilation lines

- Provision of an automatic actuation of a demineralised water pump following a trip in an operating pump

REFERENCES

[1] Wright, R. E. et al, Pipe Break Frequency Estimation for Nuclear Power Plants, NUREG-CR 4407, EG&G-2421, Idaho National Engineering Laboratories, Idaho Falls, ID, May 1987

[2] International Atomic Energy Agency, Generic initiating events for PSA for VVER reactors. IAEA-TECDOC-749, June 1994

[3] International Atomic Energy Agency, Survey of ranges of component reliability data for use in probabilistic safety assessment. IAEA-TECDOC-508, Vienna, 1989

[4] Analysis of Core Damage Frequency: Internal Events Methodology, NUREG/CR-4550, SAND86-2084, January 1990