Level 1 Shutdown and Low Power Operation of Mochovce NPP, Unit 1, Slovakia

Peter Halada, Ivan Čillík, Tibor Stojka, Miroslav Kuzma, Ján Procháška, Šuboš Vrtík
“VUJE, Inc.” Institute, Okružná 5, 91864 Trnava, Slovak Republic
haladap@vuje.sk

ABSTRACT

The paper presents general approach, used methods and form of documentation of the results that have been applied within the shutdown and low power PSA (SPSA) study for Mochovce NPP, Unit 1, Slovakia. The SPSA project was realized by VUJE Trnava Inc., Slovakia in 2001-2002 years.

The Level 1 SPSA study for Mochovce NPP Unit 1 covers internal events as well as internal (fires, floods and heavy load drop) and external (aircraft crash, extreme meteorological conditions, seismic event and influence of surrounding industry) hazards.

Mochovce NPP consists of two operating units equipped with VVER 440/V213 reactors safety upgraded before construction finishing and operation start. 87 safety measures based on VVER 440 operational experience and international mission insights were implemented to enhance its operational and nuclear safety.

The SPSA relates to full power PSA (FPSA) as a continuation of the effort to create a harmonized level 1 PSA model for all operational modes of the plant with the goal to use it for further purposes as follows:

- Real Time Risk Monitor,
- Maintenance Optimization,
- Technical Specifications Optimization,
- Living PSA.

Keywords: Probabilistic Safety Assessment (PSA), Low Power and Shutdown PSA (SPSA), Reliability Parameters

1 INTRODUCTION

The Mochovce NPP consists of two sets of twin units of VVER-440/213 reactors. Units 1 and 2 are operated since 1999 year and units 3 and 4 are under the construction. Before the units 1 and 2 start up 87 safety measures were implemented to enhance the plant operational and nuclear safety. The SPSA project of Mochovce units 1 and 2 was realized by VUJE Trnava, Inc., during the years 2001 and 2002 after the FPSA finishing.

Both FPSA and SPSA studies include internal events, internal (fires, floods and heavy load drop) and external (seismic event, air craft crash, extreme meteorological conditions and influence of external industry) hazards. Finishing the SPSA model an integrated harmonized model was created covering all operational modes. This integrated model has been used for further applications as living PSA and Safety Monitor based on SCIENTECH product version 3.5.
2 METHODOLOGY

The methodology used for SPSA performance was based in general on correspondent IAEA TECDOCs and U.S.NRC NUREGs. Reliability parameters and initiating event frequencies were based mainly on J. Bohunice V2 plant specific data, generic and supplier data of new systems and components implemented within realized safety measures (SM). Best estimate thermal-hydraulic (T-H) analyses performed for symptom based emergency operating procedures (SBEOP) and safety measures therein were used for event tree (ET) modeling and human factor analysis (HRA). Additional specific scenarios identified during event tree elaboration were calculated to support plant response modeling.

Possible recovery actions, that weren’t covered by operational procedures, were not considered within the model development to be able to evaluate the influence of safety measures consequently realized and implemented on the plant. So the SPSA results discussion was focused mainly on the possible recovery actions to restore lost systems or trains or to restart pumps or diesel generators with the aim to pay special attention to the implementation of correspondent operational procedures.

The whole project consisted of the following main tasks:
- Identification and grouping of plant operational states (POS) during low power and shutdown states of the plant,
- Identification and grouping of initiating events,
- Fault tree modeling of selected front line and auxiliary systems,
- Thermal-hydraulic analyses for selected scenarios,
- Event tree modeling,
- PSA data acquisition covering basic event reliability parameters and IE frequencies,
- Human reliability analysis,
- First quantification of the SPSA model,
- Analysis of internal fires, floods and heavy load drops,
- Analysis of external hazards as seismic event, aircraft crash, extreme meteorological conditions and influence of external industry,
- Final quantification and study results discussion,
- Harmonization of FPSA and SPSA.

3 IDENTIFICATION AND GROUPING OF PLANT OPERATIONAL STATES

The most important task of the SPSA, in comparison with the FPSA approach, was the identification and grouping of plant operational states (POS) during low power and shutdown states.

During low power and shutdown operational modes a large number of possible plant configurations and plant statuses can be defined characterized mainly by core parameters, system availability and general plant conditions. All those characteristics can form a very large number of different plant operational states. Therefore the main effort of this task was to reduce the number of initial possible plant configurations (Pre-POSs) to a manageable level and to group those Pre-POSs into limited representative group of POSs. The final list of identified POSs represents basic input information for the other tasks of the SPSA project.

The definition of the interface point between power operation and shutdown (or low power) operation for the purpose of development of the SPSA model is the first important step of this task. The FPSA typically models the full power operation and its variation between 100% and 55% of nominal power. The actual difference in reactor parameters,
thermal hydraulic response, decay heat removal, etc. does not change significantly between 100% power and low power levels (down to a few percents of power).

However, the power level of the reactor is not the only, nor from the safety perspective the most important characteristic to define the interface between both power and shutdown operation. Therefore, the following characteristics are considered to define the interface point between full power and shutdown operation for the purposes of SPSA:

- Reactor criticality (and/or shutdown margin);
- Decay heat level;
- Reactor Coolant System (RCS) temperature and pressure;
- Primary water level;
- Opening of the RCS;
- Status of RCS loops;
- Location of fuel;
- Availability of safety and support systems;
- System alignments;
- Status of the containment.

From the point of view of the objective of this task the most relevant attributes affecting the definition of the particular plant operational states are the status of the RCS loops, the system alignment and also the decay heat level.

The reactor criticality is an important attribute that influences the definition of the interface between the shutdown and power operational states. The point, when all control rods are inserted into the core, is one of the relevant features of shutdown state. However, it does not seem to be relevant in the definition of an interface between full power and low power states. The RCS parameters are not expected to influence substantially the models (given different initial power levels).

The above-mentioned criteria also determine the characteristics that are taken into account in the definition of preliminary plant operational states (Pre-POSs) and final plant operational states (POSs). See Table 1.

Operating instructions of Mochovce plant unit 1 define seven operating modes determined by the following primary system parameters:

- Temperature;
- Pressure;
- Power;
- Effective multiplication coefficient.

It is possible to define five different types of outages in Mochovce plant operation. Based on the operational history of Bohunice V-2 plant (1987-2000) and up to short operational history of Mochovce plant (1999) the following types of outages were defined:

- Very short outages. There are all planned or unplanned outages, lasting less than 24 hours.
- Short outages. There are all planned or unplanned outages, lasting > 24 but < 72 hours.
- Long outages. There are all planned or unplanned outages lasting more than 72 hours.
- Short refueling outages. There are the planned yearly outages for general plant maintenance in connection with partial refueling of the reactor.
- Long refueling outages. There are the planned outages for refueling of the reactor and extended plant maintenance. Long refueling outages are performed every fourth year and involve in-service inspection an extended scope of the vessel.
All above-defined outages are characterized by the reactor conditions - shut down, subcritical conditions reached by inserting of all control rod assemblies into the core or by establishing a shutdown boron concentration.

The examinational operation after start-up of Mochovce plant Unit 1 was finished on 29th January 1999. Since that time it is possible to consider the operation as a routine operation. During the short unit 1 operation history several planned and unplanned power reduction events were recorded. The main sources of operating history: operating histograms and operating reports.

Table 1: POSs description

<table>
<thead>
<tr>
<th>POS</th>
<th>Group of Pre-POSs</th>
<th>Operational Mode</th>
<th>POS duration</th>
</tr>
</thead>
<tbody>
<tr>
<td>POWER POS1</td>
<td>1-12</td>
<td>1-2</td>
<td>0-&gt;18.5 hrs</td>
</tr>
<tr>
<td>POS1</td>
<td>13-19</td>
<td>3</td>
<td>18.5-&gt;43.5 hrs</td>
</tr>
<tr>
<td>POS2</td>
<td>20-24</td>
<td>4-5</td>
<td>43.5-&gt;58.5 hrs</td>
</tr>
<tr>
<td>POS3</td>
<td>25-29</td>
<td>5</td>
<td>58.5-&gt;78 hrs</td>
</tr>
<tr>
<td>POS4</td>
<td>30-31</td>
<td>6</td>
<td>78-&gt;244 hrs</td>
</tr>
<tr>
<td>POS5S</td>
<td>32-34,43-46</td>
<td>6</td>
<td>169 hrs</td>
</tr>
<tr>
<td>POS5L</td>
<td>35-42</td>
<td>7</td>
<td>968 hrs</td>
</tr>
<tr>
<td>POS6</td>
<td>47-56</td>
<td>6</td>
<td>1381-&gt;1574 hrs</td>
</tr>
<tr>
<td>POS7</td>
<td>57-59</td>
<td>5</td>
<td>1574-&gt;1597 hrs</td>
</tr>
<tr>
<td>POS8</td>
<td>60</td>
<td>5</td>
<td>1597-&gt;1644 hrs</td>
</tr>
<tr>
<td>POS9</td>
<td>61-65</td>
<td>5</td>
<td>1644-&gt;1699 hrs</td>
</tr>
</tbody>
</table>

Reactor power is reduced from nominal to subcritical state. Primary pressure is nominal, primary temperature is approximately 282-258°C. All loops are operated.

Reactor is in subcritical state. Primary pressure is maintained at the value of 2 MPa and primary temperature reaches 140°C. All loops are operated.

Reactor is in subcritical state. Primary pressure is reduced from nominal value to 2 MPa and primary temperature is decreasing to 57-50°C. Core cooling is realized by forced circulation until all RCPs are consequently tripped. All loops are operated.

Reactor is in subcritical state and closed. Primary pressure (p_{PC}) is decreasing from 2 MPa to 0.3-0.5 MPa and primary temperature (T_{PC}) is maintained at 50-57°C. Core cooling is realized by natural circulation via two loops with one redundant (waiting).

Reactor is in subcritical state and opened. p_{PC} = atmospheric, T_{PC} = 50-57°C. Core cooling - by natural circulation via two loops with one redundant. Reactor dismantling for refueling is started. House power supply is realized via reserve 110 kV switchyard.

Reactor is in subcritical state and opened. Core cooling - by natural circulation via one loop with two redundant. After fueling core cooling is realized by natural circulation via one loop with one redundant (waiting).

Reactor is opened and fuel assemblies are taken out. Level of refueling and spent fuel storage pools is 21 m. This POS is suitable only for long term refueling outage. Reactor internals and all fuel assemblies are out of the reactor vessel.

Reactor is in subcritical state and opened. Core cooling is realized by natural circulation via one loop with one redundant (waiting). PC is ready for coolant filling up and closing.

Reactor is in subcritical state and closed. Primary temperature is maintained less than 57°C. Primary pressure is changed due to different pressure test performances.

Reactor is in subcritical state. Primary temperature is maintained at less than 57°C. Primary pressure is atmospheric. This POS represents the pressure test of the confinement - PERIS.

Reactor is in subcritical state. Primary temperature is maintained at less than 57°C. PC
<table>
<thead>
<tr>
<th>POS</th>
<th>66</th>
<th>5</th>
<th>1699-&gt;1723 hrs</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor is in subcritical state. Primary pressure is maintained on the value of 1.8-2 MPa. PC heating-up is started by consequential start-up of RCPs. Due to PC temperature increasing over reactor vessel brittle temperature the unit is shifted into operational mode 4.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>POS11</td>
<td>67-73</td>
<td>4</td>
<td>1723-&gt;1812 hrs</td>
</tr>
<tr>
<td>Reactor is in subcritical state. Primary temperature is maintained at 120-135°C. Primary pressure is changed during this POS due to different pressure test performances.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>POS12</td>
<td>74-75</td>
<td>4</td>
<td>1812-&gt;1820 hrs</td>
</tr>
<tr>
<td>Reactor is in subcritical state. Primary temperature is increased to temperature over 140°C. Primary pressure is maintained at the value of 2 MPa.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>POS13</td>
<td>76-79</td>
<td>3</td>
<td>1820-&gt;1847 hrs</td>
</tr>
<tr>
<td>Reactor is in subcritical state. Primary temperature is increased to 190-200°C. Primary pressure is increased from 2 MPa up to 12.26 MPa (nominal). Nitrogen cushion in pressurizer is replaced by steam.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>POS14</td>
<td>80-82</td>
<td>2</td>
<td>1847-&gt;1916 hrs</td>
</tr>
<tr>
<td>Reactor is in critical state. Primary temperature is increased up to 260 oC. Primary pressure is nominal (12.26 MPa). Physical start up is begun. Minimal critical power of the reactor is reached by boron concentration decreasing.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>POS15</td>
<td>83-92</td>
<td>2,1</td>
<td>1916-&gt;2012 hrs</td>
</tr>
<tr>
<td>Reactor power consequently arises from minimal critical power up to 20%. $p_{PC} = \text{nominal}$, $T_{PC} = 260^\circ\text{C}$. Boric acid concentration reaches critical value. Test of control rod assembly connections is performed. Power supply transition from reserve source to 400 kV switchyard.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>POWER POS2</td>
<td>93-99</td>
<td>1</td>
<td>2012-&gt;2156 hrs</td>
</tr>
<tr>
<td>Reactor power consequently arises from 20% up to nominal (100%). Primary pressure is nominal; temperature increases from 260°C to nominal.</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

4 IDENTIFICATION AND GROUPING OF INITIATING EVENTS

The list of initiating events (IEs) for FPSA was taken as the basis for IEs identification. This list was examined from low power and shutdown conditions point of view and additional events typical for SPSA were added. The following specific IEs were identified based on the detail analysis of different system trains, component maintenance, operational states:

- Man induced LOCAs due to erroneous drainage of primary pipe lines,
- Secondary circuit erroneous drainages due to wrong manipulations,
- Loss of natural circulation,
- Loss of residual heat removal due to loss of system/train on secondary site, etc.

The list of IEs was modified on different conditions of each of the 15 identified POSs. Man induced LOCAs due to erroneous drainage of primary pipelines and secondary circuit erroneous drainages due to wrong manipulation create dominant contributors during POSs 4, 5 and 6 when shutdown maintenance of the plant systems is performed.

5 FAULT AND EVENT TREE MODELING

System reliability modeling based on the fault tree approach was done according to different configurations of systems due to the applied maintenance strategy during reactor shutdown and due to different test performances during low power and shutdown states.
within the defined POSs. All relevant front line and auxiliary systems used during low power and shutdown states for residual heat removal or IEs mitigation were modeled.

For each POS the list of IEs was created based on the plant systems configuration and their availability. The plant response on each IE was modeled using the event tree approach according to actual system availabilities. The important role in the event tree modeling played an symptom based operational procedures mainly in the case of above mentioned shutdown specific IEs using different procedures to mitigate IE in different POSs based on the availability of system.

Special attention was payed to the results of previously performed and defined and performed specific T-H analyses influencing modeling of the plant response on IE.

6 SPSA DATA AND IE FREQUENCIES

FPSA data and IE frequencies were used as basis for SPSA. IE frequencies, relating to common IEs of FPSA and SPSA, were recalculated according to POS time duration considering that all operational modes represent one year. Frequencies of some rare events were stated and calculated based on the engineering or expert judgment, analysis of generic data or application of fault tree analysis.

Reliability parameters of basic event, data of unavailability due to maintenance and tests and common cause failure modeling were used the same as for FPSA. New data of components or systems used for mitigation of IEs during low power and shutdown were gained from generic databases or based on data presented by new component and system suppliers.

Common cause failure modeling was realized automatically by the PSA code - Risk Spectrum Professional.

7 HUMAN RELIABILITY ANALYSIS

The human reliability analysis (HRA) task was divided into following three steps:

- Human errors identification and description;
- Human errors qualitative analysis; and
- Human error probabilities calculation.

Basic methods applied in HRA were SLIM, THERP and ASEP. The conventional IAEA human errors classification approach was used:

Type A: Pre-initiator human errors; Type B: Human-induced initiators; Type C: Response to initiator human errors and Type D: Recovery human errors.

HRA methodology was used as well for the calculation of frequencies of the following IEs:

- Man induced LOCA due to erroneous drainage of primary pipe lines,
- Secondary circuit erroneous drainages due to wrong manipulations,
- Loss of natural circulation caused by wrong manipulation.

8 FIRST QUANTIFICATION OF THE PSA MODEL

The first quantification of the SPSA model was done after the modeling was finished. The goal of the quantification was to ensure that model is set up in an acceptable manner and that there is no logic failure. More, based on the model quantification, the truncation value was analyzed to estimate it in correct way and the sensitivity analysis was done to investigate dominant contributors of the model. The SPSA model does not include recovery actions.
Based on this analysis some corrective actions were adopted and re-analyzing of some events including human factors, initiating events, basic events, etc. to prepare the model for analysis of internal and external hazards and final quantification.

9 ANALYSIS OF INTERNAL AND EXTERNAL HAZARDS

Once the first quantification of the model was done analysis of internal and external hazards within the SPSA model started. This analysis includes detail investigation of fire and flooding sources as well as possible heavy load drops in reactor and turbine halls. Based on this detail analysis relevant IEs of internal fires, floods and heavy load drops were selected for further implementation into SPSA model.

Regarding the internal fire only fire in the turbine hall was considered as the consequence of detail analysis. Influence of internal floods is covered mainly by sources - man induced leakages or ruptures on primary and secondary sides. All of them do not lead to the flooding of any safety important components or systems.

Detailed analysis of heavy load drops in the reactor hall showed that there is no important influence on nuclear and operational safety. The same analysis in the turbine hall showed, that there are some possible events affecting operation of safety important systems in the hall (service water, residual heat removal and emergency feed water systems, etc.). Consequently the frequencies of these events were added to the original frequencies of correspondent events. Based on the analysis several recommendations to improve manipulations with heavy loads were made.

External hazards were analyzed within the SPSA in the same way as it was done in the FPSA. So only influence of seismic events and extreme meteorological conditions were considered.

10 FINAL QUANTIFICATION OF THE MODEL

After the implementation of internal and external hazards the final quantification of the model started. This task includes SPSA results presentation according to the Risk Spectrum Professional code menu. It covers not only presentation of final results as total and different IE and POS core damage frequencies, minimal cut sets, uncertainties, importance and sensitivity analyses of basic events, attributes, CCF groups and parameters, etc., but as well analysis of minimal cut sets by PSA specialists to make recommendations for improvement of the results. This item covers as well recommendations for implementation of possible recovery actions into the plant operational procedures as well as for implementation of shutdown emergency operating procedures to improve reliability of human intervention during shutdown POSs. Below are presented the main results of the SPSA (see Table 2,3).

Table 2: Percentage contribution of most important POSs to the shutdown CDF (including external hazards, shutdown CDF=1.66E-04/year)

<table>
<thead>
<tr>
<th>No.</th>
<th>POS</th>
<th>% of CDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>POS6</td>
<td>39.30%</td>
</tr>
<tr>
<td>2</td>
<td>POS4</td>
<td>34.13%</td>
</tr>
<tr>
<td>3</td>
<td>POS5S</td>
<td>15.63%</td>
</tr>
<tr>
<td>4</td>
<td>POS7</td>
<td>3.62%</td>
</tr>
<tr>
<td>5</td>
<td>POS3</td>
<td>2.97%</td>
</tr>
<tr>
<td>6</td>
<td>POS11</td>
<td>1.36%</td>
</tr>
<tr>
<td>7</td>
<td>POS9</td>
<td>1.26%</td>
</tr>
<tr>
<td>8</td>
<td>POS8</td>
<td>1.07%</td>
</tr>
<tr>
<td>9</td>
<td>Others POSs</td>
<td>0.65%</td>
</tr>
</tbody>
</table>
Table 3: Percentage contribution of initiating events to the shutdown CDF (including external hazards, shutdown CDF=1.66E-04/year)

<table>
<thead>
<tr>
<th>No.</th>
<th>IE</th>
<th>Description</th>
<th>% of CDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>L(MI)</td>
<td>Man induced LOCA</td>
<td>44.11%</td>
</tr>
<tr>
<td>2</td>
<td>LOSW(ALL)</td>
<td>Loss of all service water systems (2 out of 3)</td>
<td>22.36%</td>
</tr>
<tr>
<td>3</td>
<td>LOP</td>
<td>Loss of offsite power</td>
<td>17.61%</td>
</tr>
<tr>
<td>4</td>
<td>LVBB</td>
<td>Loss of vital operational 6kV busbar</td>
<td>3.64%</td>
</tr>
<tr>
<td>5</td>
<td>LOSW(OP)</td>
<td>Loss of working service water system</td>
<td>3.06%</td>
</tr>
<tr>
<td>6</td>
<td>COVPR</td>
<td>Cold overpressurization</td>
<td>2.63%</td>
</tr>
<tr>
<td>7</td>
<td>LNC(GP)</td>
<td>Loss of natural circulation - gas penetration</td>
<td>2.45%</td>
</tr>
<tr>
<td>8</td>
<td>ECW</td>
<td>Extremely cold weather</td>
<td>1.21%</td>
</tr>
<tr>
<td>9</td>
<td>SE</td>
<td>Seismic event</td>
<td>0.71%</td>
</tr>
<tr>
<td>10</td>
<td>Others</td>
<td>Sum of others initiating events</td>
<td>2.23%</td>
</tr>
</tbody>
</table>

11 HARMONIZATION OF FPSA AND SPSA

Harmonization of FPSA and SPSA have been started after finishing of the SPSA model final quantification. The SPSA relates to FPSA as a continuation of the effort to create harmonized level 1 PSA model for all operational modes of the plant with the goal to use it for further purposes as follows:

- Real Time Risk Monitor,
- Maintenance Optimization,
- Technical Specifications Optimization,
- Living PSA, etc.

Last year (2003) the work to create the safety monitor based on the harmonized PSA model using SCIENTECH Safety Monitor version 3.5 was finished.

12 CONCLUSIONS

Based on the SPSA model results several recommendations were made to improve plant safety. They mainly relate to the problem of shutdown emergency operating procedures and recovery actions. It means that possible recovery actions have to be implemented, as for example, restoration of service water system after its loss. Within the realization of the safety monitor based on the SPSA model some improvement of the model is required. The plant operational period is still rather short, it is supposed to upgrade the model after five years of plant operation. It means that more plant specific data will be implemented, wider statistic of planned outages will be used for more precise definition of plant operational states, actual recovery actions will be implemented.

REFERENCES