Analysis of Core Damage Frequency:

Nuclear Power Plant Dukovany,
WER/440Y-213 Unit 1, Internal Events

Volume 1: Main Report

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1.0 EXECUTIVE SUMMARY

This report presents the final results from the Level 1 probabilistic safety assessment (PSA) for the Dukovany VVER/440 V-213 nuclear power plant, Unit 1. Section 1.1 describes the objectives of this study. Section 1.2 discusses the approach that was used for completing the Dukovany PSA. Section 1.3 summarizes the results of the PSA. Section 1.4 provides a comparison of the results of the Dukovany PSA with the results of other PSAs for different types of reactors worldwide. Section 1.5 summarizes the conclusions of the Dukovany PSA.

1.1 OBJECTIVES

This project is part of a larger effort funded by the US Department of Energy (US DOE), the International Atomic Energy Agency (IAEA) and the Czech Republic State Office for Nuclear Safety (SUJB) to enhance the operational safety at VVER/440 V-213 nuclear power plants.

As the application of "state-of-the-art" PSA techniques had not yet been applied to a VVER/440 V-213 reactor design, a major goal of this project was to develop a Level 1 PSA that was equivalent to current Western Standards. The results of this study will be used as the basis for developing a real-time operational safety monitoring system for VVER/440 V-213 reactors. This real-time model will be used as part of a one year risk management pilot study at Dukovany to support day-to-day operational and maintenance planning activities. In addition, PSA models and results will be modified and used to evaluate the risk level at the Bohunice VVER/440 V-213 Unit 2 in Slovakia as part of a related effort.

The major objective of the Dukovany PSA was to provide an understanding of the types of risks (e.g., from hardware, human, or procedural deficiencies) which are most important to consider for VVER/440 V-213 reactors in general, and for Dukovany in particular. As Dukovany is in the process of upgrading the plant to conform to Western design and operating standards, the results of the PSA will provide a more detailed understanding of the types of risks that are important to consider in this process. The PSA can also be used as a living model to perform trade-off assessments, and to compare the impact of various design or procedural upgrades to enhance the plant safety.

1.2 APPROACH

A standard Level 1 PSA approach formed the basis for this analysis. The scope of the Dukovany PSA was internal initiating events, including loss of offsite power. The small-event tree/large fault tree approach was used to define accident sequences and the Integrated Reliability and Risk Assessment System (JRRAS) Version 5.0 was used for PSA quantification and uncertainty analysis. The PSA was performed by SAIC, in cooperation with Nuclear Research Institute (NRI) Rez of the Czech Republic.

As part of the basic PSA methodology that was used, five areas merit special comment:

- Because of the lack of existing best-estimate thermal hydraulics analyses for VVER/440 V-213 reactors, a large number of best-estimate calculations were performed for this study using the RELAP-5 computer code in order to define success criteria for plant systems and estimate timing for important operator actions. As a result, the success criteria used in this analysis are substantially different than what was assumed in previous VVER/440 V-213 analyses for certain types of accident sequences. The most notable changes are related to the plant response to interfacing LOCAs and unrecovered seal LOCAs through the reactor coolant pump seals.
Dukovany plant-specific raw data were used to estimate component failure rates for pumps, diesel generators, and train unavailability due to test and maintenance. In all other cases, generic data was used. The generic database that was developed for the Dukovany PSA consists of information from more than 40 relevant international data sources was aggregated to develop generic data estimates for specific basic events.

Since common cause failure data is not available for Dukovany specifically, or for VVER reactors in general, the generic alpha factor approach was used to evaluate common cause failure basic events. The common cause analysis was based on the most recent research efforts of the Electric Power Research Institute (EPRI), the US Nuclear Regulatory Commission (NRC) and IAEA.

A detailed human reliability analysis (HRA) was performed as part of this effort. The HRA specifically addresses differences between Dukovany and U.S. nuclear power plants with respect to emergency operating procedures (EOPs) and operator training. Current Dukovany procedures do not address the spectrum of actions identified in the PSA and use a narrative text format rather than columnar procedural format. Also, this plant does not have on-site, Dukovany-specific simulator training for its operators.

Anticipated transients without scram (ATWS) were not considered as part of the Dukovany PSA due to the lack of adequate best estimate thermal hydraulics analysis. However, based on the final quantified PSA results and knowledge of the failure mechanisms of the reactor protection system (RPS), ATWS is not expected to be a large contributor to the current level of core damage frequency at Dukovany.

Due to lack of information regarding the design of VVER/440 V-213 reactors (e.g., design basis information, equipment vendor information, generic and common cause failure data, lack of complete operating procedures, etc.), there were a large number of assumptions that have been used throughout the analysis, however, only a few of these assumptions had any significant impact on the overall results. In order to assess input parameter and modeling uncertainties, a Simple Monte Carlo uncertainty analysis was performed using the cutsets associated with the Dukovany total core damage frequency.

1.3 RESULTS

The total mean core damage frequency for Dukovany Unit 1 is estimated to be about 1.7E-04 per reactor year. A Simple Monte Carlo uncertainty analysis was performed to estimate the uncertainty of the results. The corresponding statistics are as follows:

- Mean Value 1.7E-4
- Median Value 9.6E-5
- 5th Percentile 3.2E-5
- 95th Percentile 4.6E-4

Of the 118 accident sequences considered as part of this study, there were 16 sequences with a frequency of greater than 1E-06, which contributed more than 9% of the Dukovany total CDF. The top 13 sequences are listed in Table ES-1 in decreasing order of frequency.

A total of twenty different initiating events were considered as part of this study. From an initiating event point of view, the Dukovany core damage frequency (CDF) is dominated by small loss of coolant accidents (LOCAs) of category L1 (10-20 min) and L2 (20-70 min), secondary pipe breaks outside confinement (T3 initiator), and loss of offsite power events (T11 initiator). Together, these four initiators account for 79% of the total CDF. Figure ES-1 provides a breakdown of the total CDF results by initiating event, including a summary of the calculated CDF for each initiator.
Table ES-1: Contribution of Important Accident Sequences to Dukovany CDF

<table>
<thead>
<tr>
<th>Sequence</th>
<th>Freq.</th>
<th>% Total CDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. L1 Small LOCA (10-20mm) with operator failure to realign the LP ECCS pump suction to prevent diversion of water from the sump.</td>
<td>4.40E-05</td>
<td>32.3%</td>
</tr>
<tr>
<td>2. Main steam collector or steamline break outside confinement (T3 initiator) with failure of all secondary cooling systems and feed-and-bleed.</td>
<td>1.95E-05</td>
<td>14.3%</td>
</tr>
<tr>
<td>3. L2 Small LOCA (20-70mm) with operator failure to realign the LP ECCS pump suction to prevent diversion of water from the sump.</td>
<td>1.10E-05</td>
<td>8.1%</td>
</tr>
<tr>
<td>4. Loss of offsite power (T11 initiator) with failure of all secondary cooling systems and feed-and-bleed.</td>
<td>1.02E-05</td>
<td>7.4%</td>
</tr>
<tr>
<td>5. Loss of offsite power (T11 initiator) with failure to provide cooling to the RCP seals, failure to recover cooling within 2 hours, and operator failure to drain the bubbler tower into confinement before the end of ECCS injection.</td>
<td>8.18E-06</td>
<td>6.0%</td>
</tr>
<tr>
<td>6. TF10 trip causing reactor scram (T7 initiator) with failure to provide cooling to the RCP seals, failure to recover cooling within 2 hours, and operator failure to drain the bubbler tower into confinement before the end of ECCS injection.</td>
<td>5.02E-06</td>
<td>3.7%</td>
</tr>
<tr>
<td>7. Loss of offsite power (T11 initiator) with failure to provide cooling to the RCP seals, failure to recover cooling within 2 hours, and failure of high pressure injection.</td>
<td>5.01E-06</td>
<td>3.7%</td>
</tr>
<tr>
<td>8. Trip of all operating SW pumps, with failure of the standby SW pumps to automatically start, failure of the operator to recover SW cooling within 15 minutes to prevent reactor trip (T10 initiator). This condition results in an unrecoverable RCP seal LOCA and core damage.</td>
<td>4.50E-06</td>
<td>3.3%</td>
</tr>
<tr>
<td>9. L1 Small LOCA (10-20mm) with failure of high pressure recirculation, and failure to manually depressurize primary pressure and recover the use of the low pressure recirculation system.</td>
<td>3.83E-06</td>
<td>2.8%</td>
</tr>
<tr>
<td>10. Feedwater collector rupture (T5 initiator) with failure of all secondary cooling systems and feed-and-bleed.</td>
<td>3.71E-06</td>
<td>2.7%</td>
</tr>
<tr>
<td>11. Loss of offsite power (T11 initiator) with failure to provide cooling to the RCP seals, failure to recover cooling within 2 hours, and operator failure to drain the bubbler tower into confinement before the end of ECCS injection.</td>
<td>3.51E-06</td>
<td>2.6%</td>
</tr>
<tr>
<td>12. Interfacing LOCA into the refueling pool (PLOCA initiator), failure of 2/3 trains of high pressure injection, and failure of the operator to drain the bubbler tower into the confinement sump.</td>
<td>3.07E-06</td>
<td>2.3%</td>
</tr>
<tr>
<td>13. Steam generator tube rupture (SGTR initiator), operator failure to isolate the primary loop with the damaged steam generator, operator failure to depressurize the primary system below the operating pressure of the secondary, and operator failure to drain the bubbler tower into the confinement sump to provide water for recirculation.</td>
<td>2.50E-06</td>
<td>1.8%</td>
</tr>
<tr>
<td>TOTAL:</td>
<td>1.24E-04</td>
<td>90.9%</td>
</tr>
</tbody>
</table>
Figure ES-1: Breakdown of Dukovany CDF Results by Initiating Event

<table>
<thead>
<tr>
<th>Initiating Event</th>
<th>CDF</th>
<th>(%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>L1 - Small LOCA (10-20mm)</td>
<td>4.80E-05</td>
<td>35.4%</td>
</tr>
<tr>
<td>L2 - Small LOCA (20-70mm)</td>
<td>1.37E-05</td>
<td>10.1%</td>
</tr>
<tr>
<td>L3 - Medium LOCA (70-100mm)</td>
<td>5.17E-07</td>
<td>0.4%</td>
</tr>
<tr>
<td>L4 - Medium LOCA (100-200mm)</td>
<td>2.92E-07</td>
<td>0.2%</td>
</tr>
<tr>
<td>L5 - Large LOCA (200-300mm)</td>
<td>7.12E-07</td>
<td>0.5%</td>
</tr>
<tr>
<td>L6 - Large LOCA (300-500mm)</td>
<td>9.89E-08</td>
<td>0.1%</td>
</tr>
<tr>
<td>ILOCA - Interfacing LOCA Outside the Confinement Boxes</td>
<td>5.01E-07</td>
<td>0.4%</td>
</tr>
<tr>
<td>PLOCA - Interfacing LOCA Inside the Refueling Pool</td>
<td>3.46E-06</td>
<td>2.5%</td>
</tr>
<tr>
<td>SGTR - Steam Generator Tube Rupture</td>
<td>2.50E-06</td>
<td>1.8%</td>
</tr>
<tr>
<td>T1 - General Plant Transient</td>
<td>7.30E-08</td>
<td>0.1%</td>
</tr>
<tr>
<td>T2 - Secondary Pipe Break Inside Confinement</td>
<td>3.61E-06</td>
<td>2.7%</td>
</tr>
<tr>
<td>T3 - Main Steam Collector and Steamline Rupture Outside Confinement</td>
<td>1.95E-05</td>
<td>14.4%</td>
</tr>
<tr>
<td>T4 - Loss of Main Feedwater System</td>
<td>3.39E-07</td>
<td>0.2%</td>
</tr>
<tr>
<td>T5 - Main Feedwater Collector Rupture</td>
<td>3.71E-06</td>
<td>2.7%</td>
</tr>
<tr>
<td>T6 - Main Feedwater-Tank Rupture</td>
<td>7.40E-07</td>
<td>0.5%</td>
</tr>
<tr>
<td>T7 - Intermediate Cooling Circuit TF10 Trip</td>
<td>5.43E-06</td>
<td>4.0%</td>
</tr>
<tr>
<td>T8 - Loss of Circulating Cooling Water (CCW) Pumps</td>
<td>6.69E-07</td>
<td>0.5%</td>
</tr>
<tr>
<td>T9 - Loss of Service Water Lines 40 and 60</td>
<td>2.83E-08</td>
<td>0.0%</td>
</tr>
<tr>
<td>T10 - Loss of Service Water System (Lines 20, 40 and 60)</td>
<td>4.50E-06</td>
<td>3.3%</td>
</tr>
<tr>
<td>T11 - Loss of Offsite Power</td>
<td>2.75E-05</td>
<td>20.2%</td>
</tr>
</tbody>
</table>
Figure ES-2 provides a breakdown of the Dukovany PSA results by grouping accident sequences (e.g., LOCAs, Transients, LOSP, etc.). Figure ES-2 also breaks down the LOCA and the LOSP and Transient categories by their major accident sequence contributors. As shown in the figure, there are three categories of sequences which contribute 76% of the total CDF for Dukovany: (i) Small LOCAs with operator failure to realign low pressure ECCS (40%), (ii) LOSP and Transient sequences involving failure of RCP seal cooling (18%), and (iii) Transients involving secondary pipe breaks with common cause failure of all secondary cooling systems (18%).

An analysis of the results reveals that the Dukovany CDF is dominated by the contribution of operator actions. Post-trip operator actions and recovery actions are represented in cutsets that account for 82% of the total CDF. This is attributable to the fact that several operator actions considered in the PSA were not proceduralized or lacked adequate indications and cues to allow for the operator to diagnose and implement specific actions. As the Dukovany procedural improvements identified in the PSA are implemented, the contribution of human failures is expected to decrease substantially.

The contribution of common cause failures (28%) is relatively small compared to most PSAs performed for Western reactor designs. This is probably attributable to the fact that Dukovany has relatively greater functional diversity than many Western reactor designs, so that common cause failures tend not to fail entire mitigating functions. It should be noted, however, that the current contribution of CCF events to plant CDF may be artificially small due to the very large contribution of human failures in the current results.

Each accident sequence is the sum of one or more combinations of events that lead to core damage. Each unique combination of events that lead to core damage is called a "cutset." In the Dukovany PSA there were more than 490,000 considered in the final quantification. However, a very small number of cutsets had a significant impact on the overall result. The top 3 cutsets contributed 50% of the total CDF. The top 67 cutsets contributed 75% of the total CDF. The top 3,957 cutsets contributed 95% of the total CDF.

It is often very informative to consider the top, or dominant, cutsets for the total core damage frequency ranked in order of contribution using various measures of importance. The risk reduction ratio provides an indication of how much the total CDF would decrease if the basic event never occurred (i.e., failure probability equal to 0). A large value indicates that a significant reduction in CDF is possible by improving the reliability associated with the event. The most significant risk reduction events for the Dukovany CDF are:

- Operator failure to realign the LP system during a small LOCA,
- Operator failure to initiate feed-and-bleed,
- Operator failure to drain the bubbler tower into the confinement sump,
- Failure of offsite power recovery,
- Diesel generator failures to run,
- Failure of PORV system motor-operated valves YP10S46(47),
- Common cause failure of the sump drain valves TZ21(41,61)S01, and
- Initiating events L1, T11, T3, L2, T7, T10, T2, T5

The inverse of the risk reduction ratio is the risk increase ratio. The risk increase ratio provides an indication of how much the total CDF would increase if the basic event always occurred (i.e., failure probability equal to 1). A large value indicates the importance of maintaining the reliability of the specific event and not allowing it to degrade. The top risk increase events for the Dukovany CDF include:
Figure ES-2: Breakdown of Dukovany CDF by Accident Sequence Grouping

TOTAL CDF

LOCA (6.3E-5 : 47%)

SGTR
(2.5E-6 : 2%)

LOCA/PLOCA
(4.0E-6 : 3%)

TOTAL CDF

LOSP and TRANSIENTS
(3.9E-5 : 28%)

LOSP (8.8E-5 : 20%)

LOSP and TRANSIENTS (48% of TOTAL CDF)

LOCAs (47% of TOTAL CDF)

A - LOSP and Transient Sequences with Failure of RCP Seal Cooling (2.4E-5 : 18% of total CDF)

B - Transients with Secondary Integrity Maintained - RCP Seals OK (5.5E-6 : 4% of total CDF)

C - Transients Involving Secondary Pipe Breaks - RCP Seals OK (2.4E-5 : 18% of total CDF)

D - LOSP - RCP Seals OK (1.2E-5 : 9% of total CDF)

A - Small LOCA Sequences with Operator Failure to Relight LP EACS Suction Lines (basic event OA-LPTRIP) (5.5E-5 : 40% of total CDF)

B - All Other LOCA Sequences (8.3E-6 : 6% of total CDF)
Common cause failure of service water pumps,
Common cause failure of batteries,
Common cause failure of TF20(40,60) intermediate cooling pumps to ECCS,
Common cause failure of diesel generators,
Common cause failure of sump suction valves TQ23(43,63)S01,
Common cause failure of sump drain valves TZ21(41,61)S01,
Common cause failure of TF10, intermediate cooling pumps to the reactor coolant pump seals,
Common cause failure of high pressure ECCS pumps
Common cause failure of high pressure ECCS air-operated valves on injection lines,
Operator actions associated with realigning the low pressure ECCS system during a small LOCA, and
Common cause failure of diesel generator circuit breakers.

It is interesting to note that there are two events that appear on both the risk increase and risk reduction lists: (i) operator actions associated with realigning the low pressure ECCS system during a small LOCA; and (ii) common cause failure of sump drain valves TZ21(41,61)S01. This indicates that the Dukovany CDF is particularly sensitive to these events. A complete list of the importance measures for the Dukovany total CDF is provided in Appendix E.

1.4 COMPARISON OF DUKOVANY PSA RESULTS TO THE RESULTS OF OTHER PSAS

As the Dukovany study is one of the first applications of PSA techniques to a VVER/440 V-213 reactor, a comparison was made to the results of other Level 1 PSAs for different types of reactors worldwide. One VVER/440 V-213, 8 PWRs and 6 BWRs were selected for the comparison [11-1 to 11-9]. Figure ES-3 shows the point estimate (mean) of the expected frequency of core damage for each of the individual studies, as well as the median value and associated uncertainty ranges, if provided. In all cases, the referenced results are for the case of Level 1 PSA at a full power operating state.

The current CDF for Dukovany is in the same range as for the US PWRs documented in the US Nuclear Regulatory Commission's NUREG-1150 risk document. These plants are of the same general age as the Dukovany NPP. Therefore, it appears that the CDF for the Dukovany NPP is about average for PWRs of that age. As discussed previously, operator action is a major contributor to Dukovany CDF (contributing to cutsets that represent more than 80% of the total CDF). The plant is currently taking actions to reduce the contribution of operator action to CDF. One of these actions is a program to develop Western-style symptom-based emergency operating procedures. In addition, the following section presents a set of specific recommended design and procedural changes intended to address the major contributors to CDF. If implemented, these changes will result in a significant reduction in the CDF for Dukovany (i.e., almost one order of magnitude). Following implementation of the of the suggested changes, it is anticipated that the Dukovany CDF will be closer to the lower end of CDFs for older PWRs and about average for the current population of NPPs world-wide.
Figure ES-3: Comparison of Dukovany PSA Results with Various Plants

(Based on Point Values, median, 5th and 95th percentiles)
1.5 CONCLUSIONS

Based on the results presented in the previous section, seven major accident were identified which contribute to Dukovany CDF. These major accident types with their associated CDF contributions are listed as follows:

1. L1 and L2 Small LOCAs with operator failure to re-align LP ECCS (40%)
2. Interfacing LOCA, including RCP seal LOCAs (21%)
3. Secondary pipe breaks (T3 and T5 initiators) with failure of secondary heat removal systems and feed-and-bleed (18%)
4. LOSP with RCP seals OK, failure of secondary heat removal systems and feed-and-bleed (8%)
5. All other LOCAs (7%)
6. Transients with secondary integrity maintained and RCP seals OK (4%)
7. SGTR (2%)

Due to the lack of design basis information and relatively little precedent in the application of PSA to VVER/440 V-213 reactors, there were a large number of assumptions used throughout the PSA analysis. Assumptions related to failure of the RCP seals following a loss of cooling, the potential for common cause failure of equipment in the Turbine Hall (elevation 14.7 meters) following secondary pipe breaks, and recovery following a total loss of service water have the largest impact on results.

Five major insights were derived from the Dukovany PSA. These insights address the major uncertainties in the analysis as well as some of the accident sequence results that are dominant contributors to the total CDF. Each insight is discussed separately as follows:

During loss of coolant accidents (LOCAs) where the pipe break is less than 45mm in diameter (L1 and L2), thermal hydraulics analysis performed in support of this study confirms that the primary pressure at the end of the ECCS injection phase is expected to be larger than the LP setpoint of 0.7 MPa. Due to the design of the ECCS system at Dukovany, the LP pumps will automatically start and recirculate water back to the LP tanks following the switchover to recirculation. At this point, operator action is required to manually opens valve TH20(40,60)S03 to allow flow from the LP tank to the ECCS common suction header. Failure of the operator to perform this action results in diversion of water from the sump to the LP tanks, the HP pumps cavitate and core damage occurs.

The analysis of this event confirmed that there is no procedural guidance or training for the operator to perform this action. The only clue to the operator that this action is required will be a "high LP tank level" light indication on the ECCS panel in the control room. The possibility for this accident sequence to occur could be eliminated if an interlock were installed that would automatically open valves TH20(40,60)S03 during the switchover to ECCS recirculation phase to allow flow from the LP tank to the ECCS common suction header. Elimination of this failure mode would result in a 42% reduction in Dukovany total CDF.

Following an initiating event involving a pipe break leading to a high energy release of steam or water into the Turbine Hall (elevation 14.7 meters), there is a potential for experiencing a complete loss of secondary cooling systems (e.g., MFW, AFW and EFW) due to the location of secondary equipment in the turbine building (elevation 14.7 m). As there was no available analysis to determine the expected phenomenological effects (i.e., steam blowdown, pipe whip, projectiles, etc) of this type of event, a point estimate based on preliminary analysis performed at Dukovany was
used to represent the possibility for common cause failure of the MFW, AFW and EFW systems for the T3 and T5 initiating events.

The impact of these events represents approximately 18% of the CDF for Dukovany. The uncertainty associated with this estimate, however, is very large. Dukovany is currently in the process of performing a more detailed analysis of the phenomenological effects of pipe ruptures in the Turbine Hall be performed and the results will be incorporated in the PSA at a later date.

The assumptions used to model the impact of failure of RCP seal cooling had a significant impact on the overall results (18% of total CDF). Based on information from the Russian manufacturer of the seals, the Dukovany PSA assumed that failure to provide cooling to the RCP seals will result in an interfacing LOCA within 2 hours with leakage transported outside confinement.

The assumptions used in the Dukovany PSA were extremely conservative compared to other PSAs that have been performed for VVER/440 V-213 reactors. More recent information from the Russian manufacturer of the RCP suggests that the integrity of the seals could be maintained for a substantially longer period following a loss of cooling. In addition, the material used to pack the seals has been changed since the original design.

More detailed analysis needs to be performed to understand the expected performance of the RCP seals when cooling is lost during accident conditions and this information needs to be reflected in the PSA. The current analysis of core damage represents a conservative assessment of the impact of failure of RCP seal integrity on core damage. During the next refueling outage, Dukovany will conduct some limited scope tests of RCP seal integrity. In addition, it is anticipated that valuable data related to the RCPs and RCP seals will be collected as part of the construction and licensing of the Mochovce VVER/440 V-213 plants in Slovakia.

The impact of operator actions on the total CDF result is substantial. Operator actions are represented in cutsets that account for 82% of the CDF. There were several operator actions considered in this study that were not proceduralized or lack adequate indications and cues to allow for the operator to properly diagnose and implement specific actions.

The operator actions that contributed significantly to the overall CDF (in terms of % contribution of cutsets to total CDF) are listed as follows:

- Operator fails to realign the LP system during a small LOCA (42%);
- Operator fails to recover offsite power to a failed 6 kV bus (25%);
- Operator fails to drain the bubbler tower into the confinement sump (14%);
- Operator fails to initiate feed-and-bleed (11%);
- Operator fails to align the water makeup system for seal cooling (4%);
- Operator fails to recover TF10 cooling to the RCP seals (1%).

It is recognized that Dukovany is in the process of developing and implementing symptom-based emergency operating procedures. It is expected that the implementation of these procedures will have a significant impact on the total CDF result. In fact, as part of the assessment of Dukovany Technical Specifications, it was estimated that the Dukovany CDF could be expected to reduce from its current value of 1.7E-04 to a value in the range of 5.5E-05 following the implementation of symptom-based emergency operating procedures [10-1].

Thus, it is recommended that efforts be undertaken to address those actions which have the largest impact on CDF now, before the eventual implementation of symptom-based procedures.
The common cause failure analysis that was performed as part of this study utilized a set of generic alpha factors based on US experience for the quantification of common cause failure (CCF) basic events in the Dukovany logic models (as recommended in NUREG/CR-5801). This approach was used due to the lack of Dukovany-specific or VVER-specific common cause failure data.

Based on an assessment of the risk achievement worth of basic events contained in the PSA, 23 of the top 25 events were common cause failures. This result indicates that the Dukovany total CDF result is very sensitive to increases in the frequency of CCF events. Thus, it is recommended that some limited scope data collection be undertaken to validate the CCF parameters that are being used for the equipment which has the highest potential impact on CDF if the common cause failure rate were to increase.

It is recognized that CCF data collection is a labor intensive and time consuming effort. A program is already under way as part of the European Community's PHARE program to develop a regional database for VVER/440 V-213 reactors. Dukovany, Bohunice (Slovakia) and Paks (Hungary) will participate in this program. It is expected that the combined operating experience of these plants will be sufficient to develop a generic component failure database for VVER/440 V-213 reactors and some limited-scope assessment of the potential for common cause failures for critical plant components. It is recommended that any future effort to assess common cause failures focus on the equipment which has the highest potential impact on CDF if the common cause failure rate were to increase.

There were also several specific recommendations which address design modifications, procedural improvements and maintenance optimization which are discussed as part of the detailed conclusions described in Chapter 11.