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NPP TEMELIN SAFETY ANALYSIS REPORTS AND PSA STATUS

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1. INTRODUCTION

Decision on the construction of a nuclear power plant at Temelin site was made in 1980 as a result of expert site selection for 4 units with VVER-1000/320 reactors. Contract on the supply of so-called „Technical design“ from the former USSR was signed in 1982. This design included turbine hall, reactor and auxiliary buildings and diesel generator stations. Design of the balance of the plant was entirely in hands of Czech party according to the contract. The Basic Design of Temelin NPP Units 1 and 2 was completed by the Czech „Architect Designer“ company Energoprojekt (EGP) Praha in 1985. The site license was issued in 1985 and construction license in November 1986. Actual erection of the buildings was launched in February 1987. Domestic specialists analyzed and subsequently modified the original design as early as before 1989. After 1989, under new political and especially economical conditions, the demand of the Czech Republic for 4000 MW power was re-evaluated and at the same time new analyses of the design safety level were performed. In March 1993 the government of the Czech Republic decided that NPP Temelin construction will be finished with two units. The date of fuel loading into the Temelin Unit 1 is August 2000, into the Unit 2 fuel will be loaded approx. 15 months later.

This paper briefly describes the effort spent to enhance safety level of the plant considering all recommendations, integrating them into the plant design, and both deterministic and probabilistic safety assessments in the frame of Safety Analyses Reports and Probabilistic Safety Assessment Project. The approach, content, current status and selected results of both types of safety assessment are discussed in this paper.

2. NUCLEAR SAFETY LEGISLATION FRAMEWORK

The history and current status of Czech nuclear safety legislation is discussed in the „Regulatory Aspects of NPP Safety“ paper. As regards Safety Analysis Reports, both the Regulation No. 85/1976 Coll. and the Regulatory Guide No 5/88 defined the types and contents of Safety Analysis Reports required by the Czech regulatory body as a basic information required for granting three level license approval during nuclear installation construction:

- For nuclear installation site license - Site (Initial) Safety Analysis Report
- For nuclear installation construction license - Preliminary Safety Analysis Report
- For nuclear installation operation license - Final Safety Analysis Report

Regulation No. 83/1976 Coll. established that those three level types of Safety Analysis Reports shall be an integral part of the documentation for nuclear installation construction.

In January 1997, Parliament of the Czech Republic passed new Act under No. 18/1997 Coll., on Peaceful Utilization of Nuclear Energy and Ionizing Radiation (the Atomic Act) and on Amendments and Additions to related Act. Amendments to this Act also newly define the content of the documentation required for siting, construction and operation of nuclear power plants.

Based on these legal requirements a licensee application for nuclear installation siting license must be accompanied by the Site (Initial) Safety Analysis Report, the content of which shall include:

1. description and evidence of suitability of the selected site from the aspect of siting criteria for

nuclear installations and very significant radiation sources as established in a legal implementing regulation;
2. description and preliminary assessment of design conception from the aspect of requirements laid down in an implementing regulation for nuclear safety, radiation protection and emergency preparedness;
3. preliminary assessment of impact of proposed nuclear installation operation on personnel, the public and the environment;
4. proposal of conception for safe termination of operation;
5. quality assurance assessment in the process of site selection, method of quality assurance for preparatory stage of construction and quality assurance principles for linking stages.

The associated implementing regulations of the Atomic Act, in accordance with the IAEA recommendations, require that assessments within the siting licensing process should consider historically most significant phenomena registered at a given nuclear installation site and its vicinity, as well as combination of various natural phenomena, resulting from human activity. Within the siting and design, nuclear installations must be evaluated as to their resistance against the following natural phenomena and phenomena caused by human activity:

- earthquakes,
- climatic effects (wind, snow, rainfall, temperature of environment etc.),
- floods and fires,
- aircrash and flying missiles,
- explosions of industrial, military and transport means, including nuclear installations,
- releases of toxic and explosive fluids and gases.

An application for the nuclear installation construction license must be accompanied with the Preliminary Safety Analysis Report (PSAR), the content of which shall include:

1. evidence that the proposed design meets all requirements for nuclear safety, radiation protection and emergency preparedness as laid down in an implementing regulations;
2. safety analyses and analyses of the potential unauthorized handling of nuclear materials and radiation sources, and an assessment of their consequences for personnel, public and environment;
3. information on predicted lifetime of nuclear installation or very significant radiation sources;
4. assessment of nuclear waste processing and management during commissioning and operation of the installation being licensed;
5. conception of safe termination of operation and decommissioning of the installation or workplace being licensed, including disposal of nuclear waste;
6. conception for spent fuel management;
7. quality assurance assessment during construction preparation, method of quality assurance for the construction and principles of quality assurance for linking stages;
8. list of classified equipment.

The license application for the nuclear installation commissioning and operation must be accompanied, among others, with the Final Safety Analysis Report (FSAR), the content of which shall include:

1. description of changes to the original design assessed in the Preliminary Safety Analysis Report and evidence that there has been no safety level decrease of the nuclear installation;
2. supplementary and more precise evidence of nuclear safety and radiation protection measures;
3. technical specifications for the nuclear installation safe operation;
4. neutron-physics characteristics of the nuclear reactor;
5. method of radioactive waste management;
6. quality evaluation of classified equipment.

3. DETERMINISTIC NUCLEAR SAFETY ASSESSMENT

3.1 Temelin Safety Analysis Reports

3.1.1 Site Safety Analysis Report

The Temelin Site Safety Analysis Report contains the assessment of the site and its vicinity as related to the safety of the NPP. It summarizes investigations and studies related to the interaction of the NPP with its environment, and on that basis, it formulates the requirements for the NPP design. Based on this report, the regulatory body, Czechoslovak Commission for Atomic Energy (at that time), issued a statement that was a legally binding document for the site selection decision making process of the civil authority. For the Temelin NPP, this stage finished in 1985.

3.1.2 Preliminary Safety Analysis Report

The Russian technical design has become a part of the Temelin Base Design including the safety assessment prepared by Energoprojekt Praha (EGP). After several reviews by ČEZ, ŠKODA, and various Czech research institutes, this design was approved in 1986. Together with the Base Design, the Preliminary Safety Analysis Report was prepared and submitted to the regulatory body for an approval.

The PSAR contains a description of the proposed design as well as the safety analyses, with the latter furnishing proof that the requirements stipulated for nuclear safety in the preceding stage of the safety documentation have been met. A favorable assessment of the PSAR by the Regulatory Body is an essential part of the licensee's application for the construction permit. The nuclear safety analyses included in the PSAR were taken from the original Russian technical design. For the first two Temelin NPP units, the PSAR was completed in 1986.

Based on this report, the Regulatory Body (ČSKAE) granted the approval (Decree No.115/86) for issuance of the construction license with the conditional requirement to complement the PSAR by additional safety analyses. These analyses were as follows:

- Faults and accidents caused by reactivity changes.
- Faults and accidents caused by RCS pressure boundary failures.
- Faults and accidents caused by failure of the balance of plant pressure boundary.
- Faults and accidents caused by component malfunctions.
- The in-core measurement system failures.
- The core power field control algorithm of the VVER 1000 reactor.
- The spent fuel pool accidents.

3.1.2.1 International Expert Appraisals

International missions to the Temelin NPP are dated since the beginning of nineties. These missions were invited to provide an independent evaluation of the original Russian design and some other aspects of the plant construction from the standpoint of internationally adopted standards.

In 1990 upon invitation from (that time) the Czechoslovak government the IAEA organized three international expert missions:

- Mission aimed at the evaluation of the Temelin site safety (April 1990)
- Pre-OSSART mission on the plant construction practice and on the preparation of safe operation (turn April/May 1990)
- Mission focused on the safety systems, core design and safety analyses evaluation (turn June/July 1990).

These missions stated that the design of NPP Temelin, its siting and organization of construction did not show any significant deviations from the international practice. The final reports of those missions [1, 2, 3] offered some partial recommendations which should have contributed to the plant safety level enhancement. The follow-up Pre-OSSART mission took place in February 1992, assessing to what degree the 1990 recommendations [4] were considered and implemented in the construction and in the preparation for future operation.

In addition to the activities listed above, the ČEZ contracted consulting company Halliburton NUS in 1991 to perform an independent audit focused on the power plant technical concept and to verify whether the plant will be licensable with respect to standards accepted for a plant built in Western Europe or the US in the mid-1990s. The audit team concluded that the overall technical concept of Temelin is in many respects consistent with modern reactor designs used in the West. Temelin includes, or can be practically modified to include, essentially all features necessary to reflect Western nuclear power plant standards designed for the mid-1990s. Some of the initial Temelin design concepts and criteria fall short of modern practices, but these shortcomings can be removed by changes in the design. These include the addition of a new I&C system, improved fuel and core design, improvements resulting from VVER and PWR operating experience, and improvements resulting from the audit team recommendations.

Besides that, some analyses were performed by COLENCO (Switzerland) and TÜV Bayern e. V. (Germany), which specifically assessed I&C design.

Among other IAEA significant activities with respect to NPP Temelin it should be especially mentioned:

- QARAT mission focused on quality assurance (turn March/April 1994) [5]
- Consultants Meeting on the Temelin design changes at the IAEA Headquarters in Vienna (turn November/December 1994) [6]
- Mission focused at the fire protection of the plant (February 1996) [7].

A special mission of the IAEA in 1996 examined how Temelin plant has solved safety issues identified by the IAEA as generic for nuclear power plants with VVER-1000/320 type reactor [8]. The mission evaluated the innovated design, implementation of previously suggested alterations and the preparation for operation. This included the compatibility issues, i.e. compatibility of modern western technology with the original Russian design. In general, the mission concluded and highly commended that the future plant operator had spent a significant effort in improving the plant design [9]. The mission emphasized that the combination of western and eastern technology in the Temelin design was considered very carefully, indeed. In mission's opinion, in some cases such combination of western and eastern technology resulted in a pronounced improvement of the safety assurance level, compared with international practice. Another Pre-OSSART mission for Temelin NPP is planned for the beginning of 2000.

3.1.2.2 Main Design Changes and Safety Improvements

Results of the independent international reviews organized by the IAEA, proposals of Czech specialists (including the SUJB recommendations) and results of the NUS Halliburton audit were used as a basis for technical improvements which, following implementation prior commissioning, will assure that both units of NPP Temelin will reach engineering standards usual for western power plants at the end of nineties. Among a number of improvements related to the replacement of components and systems, following should be mentioned:

- replacement of I&C
- replacement of core and nuclear fuel
- replacement of the original radiation monitoring system
- replacement and supplementing of the diagnostic system
- replacement of original cables with fire-retardant and fire-resistant ones

- significant changes in the electrical design (electrical protections, addition of 2 non-safety grade dieselgenerators, increased discharge time of batteries), etc.

All significant design modifications are summarized in the „Status of Safety Improvements“ paper .

3.1.2.3 Safety Analysis Recommendations

As discussed earlier, the basis of the Temelin PSAR was initially the safety analyses undertaken from the Russian technical design (TOB AES and TOB RU), which the Czech party complemented by additional safety analyses required by the regulatory body.

The last of three IAEA missions carried out under a comprehensive IAEA program for the review of the Temelin project in 1990 made several basic recommendations on accident analyses:

- To extend the safety analyses spectrum.
- To develop a systematic categorization of the accident events and to define precise rules for the classification of these events, in terms of frequency of occurrence and consequences.
- Accident consequences should be considered not only for fuel damage, but also in terms of radiological consequences.
- For each category of accident, clear acceptance criteria must be defined.
- For each accident, it was recommended that systematic, more detailed information should be given for the assumptions, initial conditions, computer codes and analytical models utilized for the analyses.

Similarly, one of the Halliburton NUS audit safety analysis recommendations stated that in the light of changes in the original Russian core design the plant safety analyses must either be extended to satisfy new requirements or be completely redone.

3.1.3 Preliminary Safety Analysis Report Amendment and Final Safety Analysis Report

Due to above mentioned design changes, as well as requirements for the safety analyses upgrade, the decision was made to integrate a new safety assessment in the Amendment of the Preliminary Safety Analysis Report, assigning the preparation of this amendment (in fact, an entire new PSAR) to a reputable Western supplier having sufficient knowledge of methodology, QA system, and licensing experience.

Based on ČEZ discussions with regulatory body and its recommendations, the EGP as the plant „Architect Designer“ developed the PSAR Amendment, with chapters 4, 7, 15, and 18 of this SAR and many portions of other chapters being within scope of supply and responsibility of Westinghouse (new I&C and core supplier). The recommended design improvements were passed to the EGP, which incorporated them as supplements to the Basic and Detail design of the plant, in cooperation with other specialized companies and cognizant technology suppliers. All safety analyses were repeated, carefully considering all technical improvements and replacements, to complement preliminary safety documentation. These analyses were performed with advanced western computer codes to the depth and in the structure required by western standards as discussed in the Temelin NPP Safety Analyses section. The final PSAR amendment was submitted to the regulatory body for comments at the end of 1995.

Since EGP Praha and ŠKODA Praha are responsible for performing the PSAR and FSAR respectively, and Westinghouse is the supplier of the new I&C, core and associated safety analyses, many discussions concerning the contents of these safety reports have been held between ČEZ and SÚJB, which resulted in the agreement to prepare both reports in accordance with US NRC Regulatory Guide 1.70 [10]. It is evident, that this US NRC guideline could not be strictly applied for the original Russian plant design at all. In addition, the PSAR amendment (and FSAR) must also reflect national requirements related to the safety report content given by the Czech Regulatory Guide 5/1988. This resulted, compared to the standard RG 1.70 content, in a minor content modifications/extensions discussed below.

The Temelin FSAR should demonstrate that NPP Temelin required safety level is achieved (compliance with legal licensing requirements) integrating all design changes made including all IAEA and OSART recommendations. In principle, the content of the report follows RG 1.70 including individual chapters numbering. Some of these chapters were extended to address specifics of the Temelin design and national legal requirements.

In comparison with the RG 1.70, an additional Chapter 18 was included to the FSAR as equipment being supplied in the frame of I&C contract must be licensable also in the country of the supplier's origin (USA) and it was essential to consider and reflect NRC requirements for the ergonomic aspects of control rooms and MMI (e.g. NUREG 800).

The Chapter 3 was modified to integrate results of LBB concept applied to Temelin RCS piping, Chapter 15 - Safety Analyses was extended to comply both Czech RG 5/88 and USNRC RG 1.70, Chapter 17 - Quality Assurance must follow approach given by the Decree 436/1990. The Chapter 16 content follows the NUREG 1431 to the extent possible, but it is modified to address NPP Temelin design specifics.

Based upon requirements of the Atomic Act No. 18/1997, the FSAR must be submitted to the regulatory body six month prior plant commissioning (fuel loading). As the fuel loading for Unit 1 is planned for August 2000, the milestones of individual FSAR chapters completion are streamed towards the end of 1999. The Temelin FSAR content and completion schedule of individual report chapters are shown in Table 1.

3.2 Temelin NPP Safety Analyses

The Westinghouse provides the fuel, I&C, safety analyses, and emergency response guidelines for Temelin NPP. This scope permits to apply the systematic and integrated approach to nuclear safety (control, monitoring, protection) that is applied also to a Westinghouse designed plants. This approach integrates: core design, plant and core monitoring, plant and core control, protection system design, safety analyses, core and plant operating limits, and emergency response guidelines. This integrated approach enhances defense in depth, which is considered to be composed of the following echelons of defense:

- Control Systems (maintains plant parameters during normal operation)
- Alarms and Manual Control (allows the operator to observe and correct deviations from normal operation)
- Limitation Systems and Backup Control (the Temelin NPP design provides for extensive "supervisory" control that can automatically take rapid action in the event of a malfunction, thus avoiding the need for protective action that would trip the plant)
- Primary Reactor Protection System (PRPS) - the safety system of Class 1 E providing automatic protection to shutdown reactor and automatic actuation and control of emergency safeguards features (ESF)
- Diverse Protection System (DPS) - the safety system of Class 1 E providing backup protection for a postulated CCF in the PRPS (provides reactor trip, some ESF actuation and control).

As stated above, a systematic approach to safety analysis is also a systematic approach to protection system design. That is, one must:

- Define unacceptable consequences (offsite dose, DNB, fuel failure, etc.)
- Determine limits on plant operation which could lead to unacceptable consequences if exceeded
- Define required protection system functions based upon events & consequences
- Select acceptance criteria, assumptions, and methods
- Analyze complete spectrum of plant conditions for the accident scenarios considered (specific analyses of particular events are of course unnecessary if an evaluation shows the event to be bounded by another event that is analyzed

- Demonstrate by results of analyses that protection system keeps the plant safe and that consequences are consistent with accepted criteria
- Develop limiting conditions for operation & monitoring requirements
    - Protection system setpoints derived from the safety analysis
    - Plant operations maintained within limits assessed in the safety analysis

3.2.1 Accident Events Categorization

The IAEA mission focused on the safety analyses recommended to develop a systematic categorization of the accident events and to define precise rules for the classification of these events, in terms of frequency of occurrence and consequences. For Temelin NPP, the approach in this area follows US NRC Regulation Guide 1.70 and ANSI/ANS 18.2, i.e.:

Condition I: Normal operation and operational transients

Condition II: Faults of moderate frequency
- No fuel failures
- No RCS overpressurization

Condition III: Infrequent faults
- Small amount of fuel damage permitted
- Dose within small fraction of 10 CFR-100 Guidelines

Condition IV: Limiting faults
- Significant fuel failures and release of radioactive material
- Dose remains within 10 CFR-100 Guidelines

These event categorizations and acceptance criteria are also fully consistent with the IAEA Guidelines for Accident Analysis for WWER Nuclear Power Plants [11].

Event categorization for the various events specified in Regulatory Guide No 5/88 and the US NRC RG 1.70 and requirements both guides for the analysis is shown in Table 2 of this paper.

3.2.2 Safety Analyses Assumptions

Since one of the purposes of the safety analyses is to define the safe limits of operation, conservative assumptions must be used which bound expected operation. Normal plant control systems are generally neglected in the safety analysis (unless their operation would cause an event to have more severe consequences). Not only are they not designed or monitored as protection systems, but plant operation is not intended to be dependent upon their operability. Similarly, the complete range of plant and core conditions (reactivity coefficients, flow maldistributions, etc.) must be covered, including allowance for uncertainties. Otherwise, plant operation could be penalized is some parameter were found to be outside the range evaluated in the safety analyses.

3.2.3 Acceptance Criteria

Acceptance criteria for various events depend upon the expected frequency of occurrence of that event; i.e., frequent events must have a large margin of safety (low consequence), and events with severe consequences must be very infrequent. Application of the ANSI/ANS-18.2 standard for Temelin NPP led to the following acceptance criteria:

Acceptance Criteria for Condition II Events (Faults of Moderate Frequency):

1. Pressure in the Reactor Coolant System (RCS) and Main Steam System (MSS) less than 110% of their design values (taken as the set pressure for the highest safety valve).
    - Max RCS Pressure < 21.1 MPa (1.1x19.2)
    - Max MSS Pressure < 9.35 MPa (1.1x8.5)
2. The DNB ratio remains above the design limit value: 95% probability of no DNB with a 95% confidence level ("95/95 DNBR")

Acceptance Criteria for Condition III Events (Infrequent Faults):

1. Shall not cause more than a small fraction of the fuel elements to be damaged
2. The radioactive releases shall be well below 10CFR-100 requirements, or lower where Czech requirements are more stringent.

Note: In some cases, the more restrictive Condition II acceptance criteria to Condition III events were applied. For example, the 95/95 DNBR criterion was also applied to sequential loss of reactor coolant flow, total loss of forced reactor coolant flow and steam line breaks of all sizes.

Acceptance Criteria for Condition IV Events (Limiting Faults):

Event specific acceptance criteria are applied to assure core coolability and acceptable radiological releases, as for example:

- Spectrum of Steam System Piping Failures - NRC requirement is DNBR >1.45 per W3 DNB Correlation
- Loss of Coolant Accidents - peak clad temperature < 1204^{o}C (also limits on zirc-water reaction, dose, core coolability)

These acceptance criteria are consistent with those recommended in Appendix 2 of the IAEA guidelines for WWER safety analysis [11].

3.2.4 Safety Analysis Methodology

The safety analysis methodology Westinghouse applied to Temelin NPP was developed from generic Westinghouse models and methods and modified as appropriate for the Temelin NPP. In each case, computer code application was reviewed, code modifications made as needed, a Temelin-specific analysis methodology established, and analysis methodology validated for VVER-1000 application. Separate reports address related analyses, including analyses of Anticipated Transients Without Scram (ATWS) Specifics are discussed below:

Large Break LOCA Analysis (LB-LOCA):

The most advanced Westinghouse LB-LOCA methodology was applied. This methodology is accepted by the US NRC and internationally (United Kingdom, Belgium, Switzerland, etc). The LB-LOCA methodology uses the WCOBRA/TRAC state-of-the-art best estimate computer code. No computer code modifications were required.

Small Break LOCA Analysis (SB-LOCA):

The current Westinghouse SB-LOCA methodology was applied. This methodology is accepted by the US NRC and internationally (e.g. Belgium, Spain, Switzerland, Taiwan, Korea, etc). The SB-LOCA methodology uses the NOTRUMP advanced nodal & general network thermal-hydraulic computer code. No computer code modifications were required.

Transient (Non-LOCA) Analysis:

Current Westinghouse transient analysis methodology was applied, as accepted by the US NRC and internationally. The transient analysis methodology uses the LOFTRAN/TWINKLE/FACTRAN code package. For these codes, minor code modifications were needed to represent the reactor coolant system, reactor protection system, 3- and 2-loop operation, etc.

Steam Generator Tube Rupture Analysis (SGTR):

The most advanced Westinghouse methodology was applied, as accepted by the US NRC and internationally. This methodology uses the LOFTTR2 Advanced Thermal-Hydraulic Computer Code. As with the transient analysis code package, minor modifications were needed to accurately represent the Temelin design.

3.2.5 Safety Analysis Application to Temelin NPP

Early in 1994, Westinghouse, NPP Temelin, and regulatory body SUJB agreed to apply the bounding analysis approach and on the events to be analyzed. Westinghouse then completed its Temelin-specific methodology development and identified the limiting cases. Licensing safety analyses has been completed and documented. Results are as follows:

Large Break LOCA:

- Limiting case is double-ended cold leg break, Cd = 0.8, with MCPs running
- Limiting Blowdown LB-LOCA peak clad temperature for 4-loop operation is less than 1204°C
- LB-LOCA during 3-loop or 2-loop operation are less limiting than during 4-loop operation

Small Break LOCA:

- Limiting case is small break in pump suction leg
- Limiting SB-LOCA peak clad temperature is less than 500°C (late in event after accumulator discharge)
- 3-loop and 2-loop operation are less limiting than 4-loop operation

Steam Generator Tube Rupture:

- Overfill can be prevented using recovery procedures similar to those in Westinghouse generic Emergency Response Guidelines
- Radiological consequences for worst case meet both US NRC and Czech acceptance criteria

Failure of Steam Generator Internal Manifold Seal:

- Core integrity and cooling are maintained
- Substantial portion of reactor coolant may be released through SG atmospheric relief valves
- Radiological consequences are within US NRC and Czech limits for postulated accident

Transient (Non-LOCA) Analysis:

- All Condition II (faults of moderate frequency) are within the DNB Core Design Limits
- All Condition III and IV events meet their respective core limits
- Most limiting condition II event is sequential loss of reactor coolant flow without limitation system action. The minimum DNB ratio is above the 95/95 DNBR design limit value.
- All Condition II, III, and IV acceptance criteria are met

DPS Verification Analysis:

Assessment of the 21 DPS design basis events (events with expected frequencies greater than once per 1000 years) shows acceptance criteria met with margin

- Core DNB design limits not breached on a realistic assessment basis
- Peak Reactor Coolant and Main Steam System pressures are less than 110% of design

Anticipated Transients without Scram (ATWS) :

- Uncontrolled nuclear response is similar to other Westinghouse-designed PWRs with a negative moderator temperature coefficient; i.e., reactor tends to shut itself down before damage limits are exceeded.
- No core fuel rod damage is expected for any anticipated transient even with no control rod insertion whatsoever
- Maximum reactor coolant system pressure does not exceed RCS test pressure (24 MPa)

4. PROBABILISTIC SAFETY ASSESSMENT

4.1 Temelin PSA Project

The probabilistic safety assessment is widely recognized as a useful tool, complementary to the standard deterministic safety assessment, for its ability to provide insights on plant safety level from the risk point of view. The utility, ČEZ-NPP Temelin decided, following the recommendations of the Risk Audit and IAEA pre-OSART missions, that the performance of a PSA would be an integral step in the preparation of the Preliminary Safety Case. The PSA of Temelin NPP Unit 1 has been performed by the Temelin PSA project team consisting of plant PSA staff, members of ÚJV, EGP, RELKO, EQE, and other Czech consultants, under the overall guidance, responsibility and project management of Halliburton NUS Corporation (Scientech, Inc.). The overall Temelin PSA project scope and current status is indicated in Table 3.

The general purpose of the PSA project was:

- to develop an appreciation of the plant severe accident behavior;
- to understand the most likely severe accident sequences that could occur at the plant;
- to obtain a quantitative understanding of the overall frequency of core damage and fission product releases and the dominant contributors to these releases;
- to transfer PSA technology to the NPP Temelin, so that the resulting models could be updated and used by plant personnel in the future for day-to-day safety related activities.

This project covers Level 1 for both at power and non-power modes of operation, external events and the Level 2 analysis. Acts of sabotage, acts of war, and off-site consequence of fission product on public health were not evaluated. There was also the requirement to transfer PSA technology to the NPP Temelin members of the project team, so that the resulting models could be used by plant personnel in the future, both on-line (Safety Monitor™), and for the off-line (NUPRA software) evaluation of a wide range of design and operational issues. The work on the Temelin PSA began in September 1993 and it was finished in June 1996 by the completion of the Level 2 analysis. Living PSA model and development and implementation of real time risk evaluation tool - Safety Monitor™ is ongoing task.

4.1.1 Methodology

The approach used for the PSA project was given by the Temelin PSA Project Plan [12] being in line with standard PSA development procedures and recommendations [16, 27, 28]. The PSA model has been developed using small event tree/large fault tree linking methodology using the NUPRA™ code, greatly facilitating the use of the model as a living tool. The event trees are „Plant Damage State„ event trees, which have been developed with the Level 2 in mind, to give a smooth interface between Level 1 and Level 2 models.

4.1.2 Data

There are several features of the Temelin plant, which impacted the PSA data estimation task. Firstly, the plant is not operational yet, so that only sources of non plant-specific data were available. While the utility ČEZ has operating experience with other VVER reactors (Dukovany NPP), it is of a different design (VVER 440/V213), thus any data is of limited direct applicability. Secondly, the plant is an unique mixture of East European and Western design. The RCS design is Russian, and a large number of the components are of Czech or West European origin, the core and the instrumentation is supplied by Westinghouse. These features therefore require that special attention be paid to ensure the applicability of any parameter adopted for the data analysis from other sources.

4.1.3 Initiating Events

Both the VVER-specific and the non-VVER specific and generic information from various sources were examined for initiating events estimation task. The analysis starting point was a detailed generic list of IEs for WWER 1000 based on the IAEA effort in the TC Project RER/9/005 [13]. Other information was obtained from VVER operational experience of Balakovo and Kalinin NPPs as well as IAEA-PRIS outage records. PWR operating experience was considered on the basis of the IEs list developed in the EPRI list of generic IEs [14]. In case of rare events a consensus of expert judgments was used based on the revised ASEP methodology manual [17]. In addition, an FMEA of selected support systems was performed to find out potential initiating events specific to the Temelin design. In some cases, such as Interfacing LOCAs, plant specific analytical models have been developed for deriving IE frequency. The IEs taken from above mentioned sources were examined for their applicability, and grouped into several general groups based on their similarities with respect to the plant response. These subevents were grouped into 5 LOCA groups and 12 transient groups. The frequency of each IE group were estimated as the sum of the subevent frequencies.

4.1.4 System Analysis

The standard approach used in [15-17] has been adopted for the modeling of system failures.

4.1.5 Equipment Data

The Temelin equipment is provided by various Czech, Eastern Europe and Western suppliers. Therefore, for the PSA model quantification, several sources of both VVER specific and non-specific data parameters were used: Dukovany NPP data collection [32], WWER 1000 data for LHI pumps, DGs and turbine bypass valves [18], the IAEA data compilation [23], the Swedish Reliability Data Book [33], and the IREP NUREG/CR-2728 [15]. NUREG/CR-1740 [19] was a source of failure rate data for instrumentation and control components.

4.1.6 Dependent Failures

The best estimate dependent failure data has been used. The pseudo plant specific database [20] and the approach described in [21, 22] have been adopted for dominant common cause failures (DGs, pumps, MOVs). In the initial phase of the Temelin PSA, a simple Beta-factor type model was used to quantify the impact of common cause failures, and to identify those common cause component groups for which a more detailed CCF analysis is required. In the final phase of the modeling, a more detailed analysis of CCFs was performed for those CCF component groups, which were found, in the initial screening analysis, to contribute significantly to the CDF. A CCF component group was considered to be significant if its FV importance measure was in excess of 1%. The detailed analysis was carried out using the Basic Parameter Model (Alfa-factor model).

4.1.7 Human Reliability

Two basic categories of human interventions have been analyzed: pre-accident and post-accident human interventions. The THERP procedure [35] and its shortened ASEP version [34] have been used for the quantification of pre-accident events. A new methodology used in the latest individual plant examination in the US, based on decision trees, has been used for quantification of post-accident events. The draft symptomatic oriented EOPs (emergency operating procedures) incorporating information on the Westinghouse system COMPRO (Computerized Procedures) has been used at the Temelin PSA study.

4.1.8 Quality Assurance

As it was intended from the project beginning that the results of the study should be incorporated in a Living PSA, one of the most important issues was quality assurance. The safety guidelines issued by the IAEA and Decree No. 436/90 [24] issued by the national regulatory body - State Office for Nuclear Safety both require implementation of a quality assurance program for activities associated with the design and operation of nuclear power plants. Therefore, a Quality Assurance Program [25] for the performance of the PSA was developed incorporating the elements of an acceptable NUS QA Program designed to meet 10 CFR 50, Appendix B [26] requirements to the extent possible. The key features of the program involve: design and documentation control, verification, review of interim and final work products, and software control. This approach has been chosen to ensure quality and will meet future requirements of the licensing authority, the development of a living PSA and the risk management program.

4.1.9 Independent Review

One of the key elements of the QA program for the Living PSA is an independent review of the work at various stages during the course of the project, which has been performed at three levels. At the first level, work products have been reviewed within the project team internally by a suitably qualified analyst not involved in the performance of the analysis task. At the second level, Temelin personnel performed an independent review of the PSA model that meets the requirements of the IAEA PSA guidelines [27, 28]. The third level of review was the review conducted by two IAEA organized IPER reviews [29, 30]. The first IPERS review of the Temelin PSA Level 1 (internal events) model, conducted by international experts took place in April/May 1995 and the second one reviewing external events and Level 2 model in January 1996.

4.2 PSA Results and Conclusions

4.2.1 Temelin PSA Results

The primary results are a delineation of the likely frequency of core damage from events which occur when the plant is operating at power, the frequency of loss of cooling (decay heat removal capability) for a number of shutdown plant operating states, and the expected magnitude and frequency of fission product release to the environment as a consequence of core melt. The core damage frequency is determined by considering in detail the occurrence of events both internal and external to the plant which will challenge the reactor protection system and require the operation of alternative systems to perform the decay heat removal function.

The result of the fire analysis do not include an evaluation of control room fires or the impact on the installed electrical cabling, as there was insufficient final installation information at the time of performing the PSA.

The core damage frequency for power operations from internal initiating events was found 8.96E-5/yr, from internal fires 1.8E-5/yr, and from floods 2.3E-6/yr. The contribution from seismic and all other external events investigated was found well below 1.0E-7/yr. The relative contributions to the CDF from internal initiating events are summarized in Table 4.

Within the most important primary to secondary leakage category, the dominant contributor is from sequences initiated by leakage from the primary header cover into the steam generator, at 4.3E-5/yr, followed by those initiated by a single tube rupture in a steam generator, 2.3E-5/yr.

These results are conservative as the steam generators fitted at Temelin are of the latest design, so they include improvements that will reduce the likelihood of header cover failures. Information on the degree of improvement was not available for the analysis reported in this study but recent information and IAEA activities on this field indicate that the frequency could be reduced by at least a factor of ten to hunderd. This would reduce the core damage frequency from this event to somewhere between 5.0E-6 to 1.0E-5 per year, and the overall core damage frequency to around 5.0E-5/yr.

The relative contributions of both internal and external initiating events are compared in Table 5 and shown in Figure 1.

4.2.2 Conclusions

The results of the core damage frequency from internal initiating events at Temelin are comparable with the results of other plants and the Temelin lies approximately within the middle of the range of results. As the design of many of the systems and the understanding of the thermal hydraulic performance of the core and RCS have been in a state of transition throughout the performance of the PSA, many conservative assumptions have been made obtaining the results in each of the tasks. Within this context, the following conclusions can be drawn from the results of the Temelin PSA.

- The dominant contribution to core damage are the sequences of events initiated by primary to secondary leakage, and in particular, steam generator header cover leakage. There are, however large conservative uncertainties surrounding this result to the design changes in the steam generator.
- There is little contribution to core damage from balance of plant failures due to the number of different ways of sustaining secondary heat removal (limitation system, auxiliary feedwater, emergency feedwater, and feed and bleed function).
- Loss of off-site power contributes less than 5 % to the total CDF, as alternative power may be available from the unit island operation, the three DGs, two non-emergency DGs, or cross connection to the other unit if it is operational. This relative contribution would rise to about 10% if the header cover failure frequency is reduced by an order of magnitude as recommended by IAEA.
- The contribution of anticipated transients with failure to scram (ATWS) to core damage is small, of the order of 3%. This is because even if the rods fail to insert, following the generation of a scram signal, there is adequate overpressure protection and a supply of feedwater from the AFW and EFW systems.
- The seismic hazard frequency at the Temelin site is very low, so the contribution to the core damage frequency from seismic events at power and shutdown is very much less than 1 percent.
- The contribution to core damage from fire and flooding events is on the order of 10 percent of that from internal events as the result of the dispersed layout and good separation of the safety-related equipment.
- The contribution to core damage frequency from external events other then internal fires, floods and seismic events was found negligible (well below 1E-7/yr) or having no impact on the plant safety.

4.3 Further PSA Activities

As the plant is currently under construction with many safety improvements being implemented into design, the information required for the PSA have been in a state of transition throughout the performance of the analysis and through this, the results are based on a number of conservative assumptions concerning the ultimate design and future operation in some areas. Some of the information used in the analysis relating to the dominant contributors to core damage and off-site risk is based on the earlier steam generator design and some of the information in the areas of I&C, cable installation, and control rooms layout were not available at the time of the analysis. These have all been carefully documented so that they can be confirmed, corrected or amplified, as the final plant information becomes available prior commissioning. This documentation clearly delineates the information and assumption(s) made in each area of the work, and hence, enables the model to be updated as further information, and ultimately, operating experience become available.

In order to overcome the conservatism contained in the PSA, Temelin NPP plan to update all models in the 1999-2000 time frame, to ensure that each information and assumption in the PSA reflects the ultimate design and construction of the plant at the time of the plant commissioning, providing risk model that is suitable both for use in licensing-related work as well as various applications.

4.4 PSA Applications

The increasing level of plant design and operational details, which is possible to include in the PSA and the ability to relate this directly to the individual PSA model part has lead to the realization that the PSA can be used in the day-to-day risk informed operation and decision making processes at the plant. From the outset there were two clear aims for the performance of the Temelin PSA. The first aim was to ensure that the Temelin personnel were fully trained in the PSA methodology so that a full in-house capability would be developed for the future use of the PSA. The second was to produce a „live„ plant model, which could be used now, and in the future, to provide important information on plant safety and operation.

It is expected that the model will be used extensively in the future for various applications to assess the impact on safety of design changes, procedural changes, changes in the technical specifications, and changes in test and maintenance procedures and strategies based on risk minimization. Also, it is expected that the model will be used by plant training personnel to gain a better understanding of certain accident sequences, thus enhancing the training given to plant personnel. Other areas of PSA application at the plant are evaluation of accident events (precursors), outage risk management or providing historical risk profiles associated with the plant operation.

4.5 Safety Monitor

Above mentioned applications are primarily associated with calculating risk for a given plant configuration or its change. To change the risk it must eventually either change the value of the individual elements in the PSA (basic events) or the logic structure in which those elements are arranged. As calculating the risk of a new plant configuration using PSA risk model requires roughly a person days, performing a risk assessment sometimes for hundreds of equipment outages, tests, and alignments required for some applications is thus beyond reach. Also, the PSA is performed using a set of computer codes specifically designed for a very detailed analysis of the plant, requiring a comprehensive knowledge of the methodology and techniques for the performance of a PSA. Such codes are not suitable for use by non-PSA specialists.

Therefore, in order to provide a risk model for daily use by the other engineers at the plant, the PSA Project has been extended, and the risk model developed within the PSA was transferred to a real-time risk calculation software analyzing both real and scheduled plant conditions for determining the impact of plant configurations on operational risk level - NPP Temelin Safety Monitor [31].

The major purpose of the Safety Monitor™ at Temelin is to provide an on-line risk measure based on the current plant configuration, on-line maintenance or testing status, so enabling plant staff to plan and perform maintenance activities in such a way that safety is maximized, and at the same time unnecessary plant shutdown is avoided. In this software the user enters the plant status using the component nomenclature in daily usage, as well as changes in plant alignment, or tests in progress. The risk level associated with that specific configuration is then calculated within the software and displayed on a meter. The meter shows three risk operating bands, low, warning and high. At the same time a suggested allowed configuration time associated with that risk level is displayed on the screen.

It is expected that the Safety Monitor will be used extensively by the maintenance division in scheduling preventive maintenance activities, both at power and during refueling. The planner can enter specified equipment outages several weeks in advance and if the calculated risk for these hypothetical configurations is high, then equipment outages can be rearranged within the schedule until the risk is sufficiently low. Similarly for operator personnel would definitely be an advantage to be informed about a risk associated with an upcoming plant configuration prior such configuration will be entered, including recommended limiting time to be allowed for such configuration or to provide risk reduction advise for such configuration. In such manner, through the Safety Monitor, the PSA becomes a tool for active influence on operational risk level without detailed knowledge of PSA techniques and terminology, at the same time providing means to optimize safety within Technical Specifications constraints, planned maintenance activities and storing history of plant configuration changes and component outages with associated risk levels.

5. TEMELIN SAFETY EVALUATION SUMMARY

To enhance the safety level of the Temelin plant, recommendations of the independent international reviews were implemented into the design as well as into the organization of the plant construction and preparation for operation. The safety assessment of these design changes has been integrated and reflected in the corresponding Safety Analysis Reports, content of which follows internationally accepted guidelines.

All safety analyses within Safety Analysis Reports were repeated, carefully considering the technical improvements and replacements, to complement preliminary safety documentation. These analyses were performed with advanced western computer codes to the depth and in the structure required by western standards. The Temelin NPP followed a systematic approach in the functional design of the Reactor Protection System (PRPS and DPS) and related safety analyses. The reactor protection system modifications increase defense in depth and facilitate demonstrating that applicable safety limits are met for non-LOCA events. The rigorous safety analysis methodology provides assurance that LOCA and radiological limits are met. Established and accepted safety analysis methodology and acceptance criteria were applied to Temelin, meeting: US NRC and Czech Republic requirements, IAEA guidelines and recommendations.

In addition to the deterministic safety analyses, the plant specific PSA was performed within Temelin PSA Project. As many safety improvements were implemented into design throughout the performance of the analysis, the information required for the PSA has been in a state of transition and through this, the results are based on a number of conservative assumptions concerning the ultimate design and future operation in some areas. In order to ensure that each information and assumption in the PSA reflects the ultimate design and construction of the plant (including all plant modifications) at the time of the plant commissioning, Temelin NPP plan to update all models in the 1999-2000 time frame. This will provide the „living" risk model that is suitable both for use in licensing-related work as well as for various safety related applications.

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Table 1 - Table of Contents and Completion Milestones of Temelin NPP FSAR

No	Chapter	Completed	Note
1.	**INTRODUCTION AND GENERAL DESCRIPTION OF PLANT**	**9/99**	
1.1	Introduction		
1.2	General Plant Description		
1.3	Design Comparison Tables		
1.4	Identification of Agents and Contractors		
1.5	Requirements for Further Technical Information		
1.6	Material Incorporated by Reference		
1.7	Drawings and Other Detailed Information		
1.8	Conformance to Regulatory Body Guides		
2.	**SITE CHARACTERISTICS**	**8/99**	
2.1	Geography and Demography		
2.2	Nearby Industrial, Transportation, and Military Facilities		
2.3	Meteorology		
2.4	Hydrologic Engineering		
2.5	Geology, Seismology, and Geotechnical Engineering		
3.	**DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS**	**10/99**	
3.1	Conformance with Regulatory Body General Design Criteria		
3.2	Classification of Structures, Components and Systems		
3.3	Wind and Tornado loadings		
3.4	Water Level (Flood) Design		
3.5	Missile Protection		
3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping		
3.7	Seismic Design		
3.8	Design of Category I Structures		
3.9	Mechanical Systems and Components		
3.10	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment		
3.11	Environmental Design of Mechanical and Electrical Equipment		
4.	**REACTOR**	**8/99**	**WEC**
4.1	Summary Description		
4.2	Fuel System Design		
4.3	Nuclear Design		
4.4	Thermal and Hydraulic Design		
4.5	Reactor Materials		
4.6	Functional Design of Reactivity Control Systems		
5.	**REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS**	**6/99**	
5.1	Summary Description		
5.2	Integrity of Reactor Coolant Pressure Boundary		
5.3	Reactor Vessels		
5.4	Component and Subsystem Design		
5.5	Auxiliary Systems of Reactor Coolant Systems		
6.	**ENGINEERED SAFETY FEATURES**	**9/99**	
6.1	Engineered Safety Feature Materials		
6.2	Containment Systems		
6.3	Emergency Core Cooling System		
6.4	Control Room Habitability Systems		
6.5	Fission Product Removal and Control Systems		
6.6	Inservice Inspections of Safety Related Components		
6.7	Not applicable for Temelin (BWR)		
6.8	RCS Steam-Water Mixture Venting System		
6.9	Steam Generator Emergency Feedwater System		
7.	**INSTRUMENTATION AND CONTROL**	**10/99**	**WEC**
7.1	Introduction		

No	Chapter	Completed	Note
7.2	Reactor Trip System		
7.3	Engineered Safety Feature Systems		
7.4	Systems required for Safe Shutdown		
7.5	Safety Related Display Instrumentation		
7.6	All Other Instrumentation Systems Required for Safety		
7.7	Control Systems Not Required for Safety		
8.	**ELECTRIC POWER**	**9/99**	
8.1	Introduction		
8.2	Offsite Power Systems		
8.3	Onsite Power Systems		
9.	**AUXILIARY SYSTEMS**	**8/99**	
9.1	Fuel Storage and Handling		
9.2	Water Systems		
9.3	Process Auxiliaries		
9.4	Air Conditioning, Heating, Cooling and Ventilation Systems		
9.5	Other Auxiliary Systems		
9.6	Plant offsite central heat supply		
10.	**STEAM AND POWER CONVERSION SYSTEM**	**6/99**	
10.1	Summary Description		
10.2	Turbine-Generator		
10.3	Main Steam Supply System		
10.4	Other Equipment of Steam and Power Conversion System		
11.	**RADIOACTIVE WASTE MANAGEMENT**	**9/99**	
11.1	Source Terms		
11.2	Liquid Waste Management Systems		
11.3	Gaseous Waste Management Systems		
11.4	Solid Waste Management Systems		
11.5	Process and Effluent Radiological Monitoring and Sampling Systems		
12.	**RADIATION PROTECTION**	**9/99**	
12.1	Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)		
12.2	Radiation Sources		
12.3	Radiation Protection Design Features		
12.4	Dose Assessment		
12.5	Health Physics Program		
13.	**CONDUCT OF OPERATIONS**	**9/99**	
13.1	Organizational Structure of Applicant		
13.2	Training		
13.3	Emergency Planning		
13.4	Review and Audit		
13.5	Plant Procedures		
13.6	Industrial Security		
14.	**INITIAL TEST PROGRAM**	**9/99**	
14.1	N/A for FSAR		
14.2	Specific Information To Be Included in Final Safety Analysis Reports		
15.	**ACCIDENT ANALYSES**	**8/99**	**WEC**
15.X	Evaluation of Individual Initiating Events		
16.	**TECHNICAL SPECIFICATIONS**	**9/99**	
16.1	Preliminary Technical Specifications (PSAR)		
16.2	Proposed Final Technical Specifications		
17.	**QUALITY ASSURANCE**	**10/99**	
17.1	Quality Assurance During Design and Construction		
17.2	Quality Assurance During the Operation Phases		
18.	**ERGONOMIC AND ENGINEERING CRITERIA**	**9/99**	**WEC**

Table 2 - Comparison of Safety Analyses Required by the Czech Regulatory Guide 5/88 and US NRC RG 1.70 (Temelin FSAR Chapter 15)

Chapter	ACCIDENT/CASE DESCRIPTION	SÚJB RG 5/88	WEC Approach	ANSI Class
15.0	**Set of the complete input data used for accident analyses**			
15.1	**Increase in heat removal by the secondary system**			
15.1.1	Feedwater system malfunctions causing a reduction in feedwater temperature	Y	Analysis	II
	I. Full opening of the bypass valve of a regenerative heat exchanger (RHE)	Y	Analysis	
	II. Trip of all RHEs	Y	Analysis	
	III. Trip n/3 legs of heat exchangers (n = 1, 2, 3)	Y	EGP	
15.1.2	Feedwater system malfunctions causing an increase in feedwater flow	Y	Analysis	II
	a) Accidental opening of FCV at full power	N	Analysis	
	- automatic rod control	N	Analysis	
	- manual rod control	N	Analysis	
	b) Accidental opening of FCV at zero power	N	Analysis	
15.1.3	Steam pressure regulator malfunction or failure that results in increasing steam flow	Y	Analysis	II
	a) Minimum feedback + automatic rod control	N	Analysis	
	b) Minimum feedback + manual rod control	N	Analysis	
	c) Maximum feedback + automatic rod control	N	Analysis	
	d) Maximum feedback + manual rod control	N	Analysis	
15.1.4	Inadvertent opening of SG relief or safety valve causing a depressurization of the MSS	Y	Anal/Eval	II
	a) Inadvertent opening of 1 ADV	Y	Anal/Eval	
	b) Inadvertent opening and stuck open SGSV	Y	Evaluation	
	c) Inadvertent opening of turbine bypass valve	Y	Anal/Eval	
15.1.5	Spectrum of main steam line breaks	Y	Anal/Eval	III & IV
	a) Rupture of steam line inside containment	Y	Anal/Eval	
	b) Rupture of steam line outside containment	Y	Anal/Eval	
	c) Rupture of main steam header	Y	Anal/Eval	
15.2	**Decrease in heat removal by the secondary system**			
15.2.1	Steam pressure regulator malfunction or failure that results in decreasing steam flow	N/A	Evaluation	II
15.2.2	Loss of external electrical load	Y	Evaluation	II
15.2.3	Turbine trip (TG stop valve closure)	Y	Analysis	II
	a) Minimum feedback with pressure control	N	Analysis	
	b) Minimum feedback without pressure control	N	Analysis	
	c) Maximum feedback with pressure control	N	Analysis	
	d) Maximum feedback without pressure control	N	Analysis	
15.2.4	Inadvertent closure of the MSIVs	Y	Analysis	II
15.2.5	Loss of condenser vacuum	N/A	Evaluation	II
	a) Trip 1/2 condenser cooling water pumps	Y	Evaluation	
	b) Trip 2/2 condenser cooling water pumps	Y	Evaluation	
15.2.6	Loss of onsite and offsite AC power	Y	Analysis	II
	a) Main TG breaker trip	Y	Evaluation	
	b) Loss of onsite AC power	Y	Evaluation	
15.2.7	Loss of normal feedwater flow	Y	Anal/Eval	II
	a) LONF with offsite power, with No MCP Trip	N	Analysis	
	b) LONF with offsite power, with MCP Trip	N	Analysis	
	c) LONF without offsite power	Y	Analysis	
	d) Loss of 1 MFW pump	Y	Evaluation	
	e) Loss of 2 MFW pumps and failure of AFW pump	Y	Evaluation	
	f) Closure of isolation valve on main feedwater line	Y	Evaluation	
	g) Loss of condensate pumps (different combinations)	Y	Evaluation	

Chapter	ACCIDENT/CASE DESCRIPTION	SÚJB RG 5/88	WEC Approach	ANSI Class
	h) Loss of essential cooling water pumps	Y	Evaluation	
15.2.8	Feedwater line breaks	Y	Anal/Eval	IV
	a) Main feedwater line break, offsite power available	N	Analysis	
	b) Main feedwater line break, offsite power unavailable	N	Analysis	
	c) Break downstream and upstream check valves	Y	Evaluation	
	d) Main feedwater header rupture	Y	Evaluation	
15.3	**Decrease in reactor coolant system flow rate**			
15.3.1	Single and multiple MCP trips and complete loss of forced reactor coolant flow	Y	Anal/Eval	II & III
	a) 4/4 - CLOF	Y	Analysis	III
	b) 3/3 - CLOF	Y	Analysis	III
	c) 2/2 - CLOF	Y	Analysis	III
	d) 3/4 - PLOF	Y	Evaluation	II
	e) 2/4 - PLOF	Y	Analysis	II
	f) 1/4 - PLOF	Y	Analysis	II
	g) 2/3 - PLOF	Y	Evaluation	II
	h) 1/3 - PLOF	Y	Evaluation	II
	i) 1/2 - PLOF	Y	Evaluation	II
	j) 1/4 + 3/4 - SLOF	N	Analysis	III
	k) 1/3 + 2/3 - SLOF	N	Analysis	III
	l) 1/2 + 1/2 - SLOF	N	Analysis	III
15.3.2	Not applicable for NPP Temelin	-	-	-
15.3.3	Main coolant pump shaft seizure (locked rotor)	Y	Analysis	IV
	a) 1/4	Y	Analysis	
	b) 1/3	Y	Analysis	
	c) 1/2	Y	Analysis	
15.3.4	Main coolant pump shaft break	N/A	Analysis	IV
	a) 1/4	N	Analysis	
	b) 1/3	N	Analysis	
	c) 1/2	N	Analysis	
15.3.5	Deterioration of heat removal	Y	EGP	II
	a) Cooling down by natural circulation	N	EGP	
	b) Cooling down by coolant vaporization	N	EGP	
	c) Partial coolant flow blockage through the fuel assembly	N	EGP	
15.4	**Reactivity and power distribution anomalies**			
15.4.1	Uncontrolled control rod bank withdrawal from a subcritical or low power startup condition	Y	Anal/Eval	II
	a) Analyzed to evaluate minimum DNBR	N	Analysis	
	b) Analyzed to evaluate hot spot fuel temperature	N	Analysis	
15.4.2	Uncontrolled control rod bank withdrawal at power	Y	Analysis	II
	a) 4 Loop, 100% power, Maximum feedback	N	Analysis	
	b) 4 Loop, 100% power, Minimum feedback	N	Analysis	
	c) 4 Loop, 60% power, Maximum feedback	N	Analysis	
	d) 4 Loop, 60% power, Minimum feedback	N	Analysis	
	e) 4 Loop, 10% power, Maximum feedback	N	Analysis	
	f) 4 Loop, 10% power, Minimum feedback	N	Analysis	
	g) 3 Loop, 67% power, Maximum feedback	N	Analysis	
	h) 3 Loop, 67% power, Minimum feedback	N	Analysis	
	i) 3 Loop, 10% power, Maximum feedback	N	Analysis	
	j) 3 Loop, 10% power, Minimum feedback	N	Analysis	
	k) 2 Loop, 50% power, Maximum feedback	N	Analysis	
	l) 2 Loop, 50% power, Minimum feedback	N	Analysis	
	m) 2 Loop, 10% power, Maximum feedback	N	Analysis	
	n) 2 Loop, 10% power, Minimum feedback	N	Analysis	
15.4.3	Control rod malfunction	Y	Analysis	II
	a) One or more RCCAs within the same group dropped	Y	Analysis	

Chapter	ACCIDENT/CASE DESCRIPTION	SÚJB RG 5/88	WEC Approach	ANSI Class
	b) A dropped RCCA bank	Y	Analysis	
	c) Statically misaligned rod	Y	Analysis	
	d) Single rod withdrawal at power	Y	Analysis	
15.4.4	Startup of an inactive RCS loop at an incorrect temperature	Y	Analysis	II
	a) Transition from 3-loop to 4-Loop Operation	N	Analysis	
	b) Transition from 2-loop to 3-Loop Operation	N	Analysis	
15.4.5	Not applicable for NPP Temelin	-	-	-
15.4.6	Chemical and volume control system (TK) malfunction resulting to a RCS boron concentration decrease	Y	Anal/Eval	II
	a) Uncontrolled boron dilution in coolant	Y	Anal/Eval	
	– Boron dilution in coolant, mode 1	N	Analysis	
	– Boron dilution in coolant, mode 2	N	Analysis	
	– Boron dilution in coolant, mode 3	N	Evaluation	
	– Boron dilution in coolant, mode 4	N	Evaluation	
	b) Sudden transition to RCS charging by 60 - 70 deg C charging water	Y	Evaluation	
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper position	N/A	Analysis	III
	a) Misloading error	N	Analysis	
	b) Enrichment error	N	Analysis	
15.4.8	Spectrum of rod ejection accidents	Y	Anal/Eval	IV
	a) Hot full power, Beginning of life	N	Analysis	
	b) Hot full power, End of life	N	Analysis	
	c) Hot zero power, Beginning of life	N	Analysis	
	d) Hot zero power, End of life	N	Analysis	
15.4.9	Not applicable for NPP Temelin	-	-	-
15.5	**Increase in reactor coolant inventory**			
15.5.1	Inadvertent actuation of the ECCS during power operation	N/A	Evaluation	II
15.5.2	Chemical and volume control system malfunction that increases reactor coolant system inventory	N/A	Anal/Eval	II
	a) CVCS malfunction analyzed to evaluate minimum DNBR	N	Analysis	
	b) Inadvertent injection from standard charging system into pressurizer with water temperature 60 - 70 deg C	Y	Evaluation	
	c) Pressurizer spray valve opening and stuck open	Y	Evaluation	
15.5.3	Not applicable for NPP Temelin			
15.6	**Decrease in reactor coolant inventory**			
15.6.1	Inadvertent opening of a pressurizer safety or relief valve	Y	Anal/Eval	II & III
	a) Pressurizer safety valve inadvertent opening and stuck open	Y	Analysis	
	b) Insufficient closing of a pressurizer safety valve following right opening	Y	Evaluation	
15.6.2	Breaks at instruments line or other RCPB lines penetrating the containment	Y	Evaluation	II& III
15.6.3	Steam generator tube rupture	Y	Anal/Eval	IV
15.6.4	Steam generator internal manifold failure	Y	Anal/Eval	IV
15.6.5	Loss of coolant accidents (LOCAs)	Y	Analysis	III & IV
	a) Large Breaks	Y	Analysis	
	b) Small Breaks	Y	Analysis	
15.7	**Radioactive release from a subsystem or component**			
15.7.1	Radioactive gas waste system leak or failure	Y	SKODA	III
15.7.2	Radioactive liquid waste system leak or failure	Y	SKODA	III
15.7.3	Postulated radioactive releases due to liquid tank failures	N/A	SKODA	III
15.7.4	Design basis fuel handling accident in the containment	Y	Analysis	IV
	a) Dropped assembly impacts another assembly still in the reactor core	N	Analysis	
	b) Dropped assembly drops into the spent fuel canal, striking the floor of the canal	N	Analysis	

Chapter	ACCIDENT/CASE DESCRIPTION	SÚJB RG 5/88	WEC Approach	ANSI Class
	c) *Dropped assembly drops into the spent fuel storage pool,* striking either the floor of the pool or another fuel assembly in the spent fuel storage racks	N	Analysis	
15.7.5	Spent fuel cask drop accidents	Y	SKODA	III
15.8	**Anticipated transients without scram (ATWS)**			
15.8.1	Inadvertent control rod withdrawal	Y	Analysis	II
15.8.2	Loss of feedwater	Y	Analysis	II
15.8.3	Loss of AC power	Y	Analysis	II
15.8.4	Loss of electrical load	Y	Evaluation	II
15.8.5	Loss of condenser vacuum	Y	Evaluation	II
15.8.6	Turbine trip	Y	Analysis	II
15.8.7	Closure of main steam line isolation valves	Y	Analysis	II
15.8.8	Reactor shutdown to hot standby state by boron concentration increasing	Y	EGP	N/A
15.9	**Loss of essential cooling water pumps**	Y	EGP	II

Table 3 - Temelin PSA Scope and Status

PSA Scope and Status	
Scope:	Level 1 - internal initiating events Level 1 - external initiating events (fire, floods, seismic, others) Level 2 Living PSA (Temelin Safety Monitor™)
Operating modes:	Full power, shutdown (outages, refueling)
Current status:	Level 1 - internal IEs completed Level 1 - external IEs completed Shutdown completed Level 2 completed Safety Monitor 2.0™ ongoing

Table 4 - Summary of Internal Initiators Contribution to CDF

Initiating Events		Contribution to CDF [1/yr]	Contribution to CDF [%]
T9	SG Header Cover Leakage	4.33E-5	48.3
T8	Single SG Tube Rupture (SGTR)	2.27E-5	25.4
S2	Large LOCA	7.87E-6	8.8
S4	Small LOCA	4.01E-6	4.5
TS	ATWS	2.70E-6	3.0
T6	Loss of Offsite Power	2.68E-6	3.0
T4	Reactor Trip - Main Feedwater Unavailable	8.73E-7	1.0
S5	Very Small LOCA	1.94E-6	2.2
T1	Reactor Trip - All Systems Available	8.73E-7	1.0
S3	Medium LOCA	4.61E-7	0.5
S1	Reactor Vessel Rupture	2.71E-7	0.3
T5	Reactor Trip - Main and Auxiliary Feedwater Unavailable	2.13E-7	0.2
TV2	Loss of two VF Cooling trains VF10, 20	1.95E-7	0.2
V	Interfacing system LOCA	1.60E-7	0.2
TV	Loss of all VF Cooling trains	5.23E-9	0.0
T7	Unisolable steamline/FW line breaks	0.0	0.0

Table 5 - Summary of Relative Contributions of Internal and External Initiators to CDF

Initiating Event Group	Core Damage Frequency [1/yr]	Total CDF [1/yr]
Internal Initiating Events		8.96E-5
Primary to Secondary Leakage	6.6E-5	
LOCAs	1.5E-5	
Transients	6.3E-6	
Anticipated Transients Without Scram (ATWS)	2.7E-6	
Fire		1.8E-5
Cable Spreading Room (N02.10)	2.1E-6	
Cable Spreading Room (N02.09)	1.7E-6	
Turbine building (490/TH)	1.6E-6	
Others	1.26E-5	
Flood		2.3E-6
Turbine building (1)	9.4E-7	
Reactor Building (+6.6 m)	5.0E-7	
Reactor Building (2)	4.6E-7	
Reactor Building (3)	1.7E-7	
Reactor Building (-4.2 m)	1.6E-7	
Others	8.3E-8	

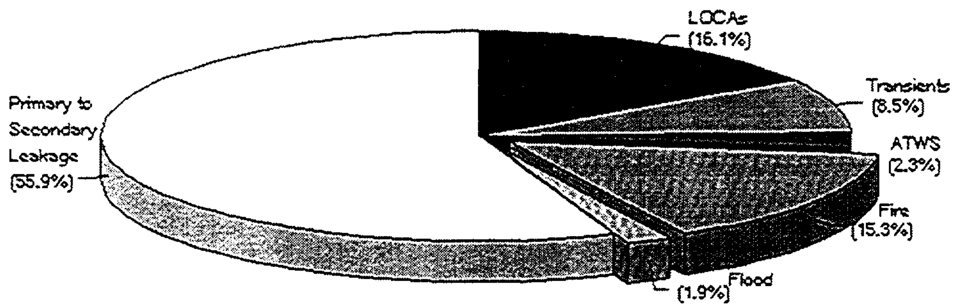

Figure 1 - Relative Contribution of Internal and External Initiators to Total CDF

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