Preliminary Safety Assessment of the Methodology and Strategy for DEC-A in APR1000

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1. Introduction

Design Extension Conditions (DECs) are defined as "postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best-estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with core melting" [1]. This paper focuses only on the DECs without significant fuel degradation, referred to as DEC-A.

In choosing the DEC-A events to be addressed in the design, the following factors should be considered together.

- the frequency of the event
- the grace period for necessary human actions
- the margins to cliff edge effects
- the radiological or environmental consequences

The list of DEC-A events were established in accordance with EUR Requirements [2], conservatively encompassing the event lists provided by the Korea [3], IAEA [4], WENRA [5] and APR1000 standard design. The finalized DEC-A events can be categorized into the groups below, resulting in a total of 13 DEC-A events.

(1) Group 1 : Anticipated Operational Occurrence (AOO) or most-frequent Design Basis Accident (DBA) 1 events combined with postulated Common Cause Failure (CCF) of redundant trains of required safety systems
(2) Group 2 : Complex or specific scenarios including CCFs of systems needed to fulfil the fundamental safety functions in normal operation.
(3) Group 3 : Combination of failures including additional failures resulted from the consequence of Postulated Initiating Event (PIE)

Table 1 describes a total of 13 DEC-A events, which will be analyzed to support the Dukovany 5&6 (DKN5&6) Preliminary Safety Analysis Report (PSAR). The design of DKN5&6 will be based on the APR1000 standard design.

Table 1. DEC-A events in PSAR Chapter 20

Section	Event
20.1.1	Anticipated Transient Without Scram (ATWS) due to Mechanical Blocking of Rods
20.1.2	Anticipated Transient Without Scram (ATWS) due to Failure of Reactor Protection System (RPS)
20.1.3	Station Blackout (SBO)
20.1.4	Total Loss of Feedwater to the Steam Generators (TLOFW)
20.1.5	Loss of Coolant Accident (LOCA) with Loss of Safety Injection
20.1.6	Uncontrolled Boron Dilution
20.1.7	Loss of Ultimate Heat Sink during Normal Operation (LOUHS)
20.1.8	Total Loss of Cooling Chain during Normal Operation
20.1.9	Loss of Spent Fuel Cooling Functions (LOSFPC) during Normal Operation
20.1.10	Multiple Steam Generator Tube Ruptures (MSGTR)
20.1.11	Main Steam Line Break (MSLB) with Consequential Steam Generator Tube Ruptures (SGTR)
20.1.12	Loss of Residual Heat Removal System (LORHR)
20.1.13	Interfacing System LOCAs (ISLOCA)

This paper presents a preliminary safety assessment for DEC-A analyses, including assumptions, acceptance criteria, event scenarios, safety and dedicated features, and mitigation strategies. This assessment provides the design basis for evaluating DEC-A events in Chapter 20 of DKN 5&6 PSAR.

2. Methodology for DEC-A Safety Analysis

2.1 Analysis Methodology for DEC-A

According to EUR requirements [2], the safety analysis for DECs shall rely on best-estimate methodologies. Therefore, normal operating conditions (initial and boundary condition) and normal plant design values such as nominal setpoints for overpressure protection devices and engineered safety features are assumed to apply. Additional single failure is not assumed and off-site power is available during the transient.

The safety analysis shall take into account the uncertainties and demonstrate by uncertainty and/or sensitivity analysis that core damage can be prevented with an adequate level of confidence and there is adequate margin with regard to cliff-edge effects.

Operator actions are not assumed during the first 30 minutes in the Main Control Room (MCR) or 60 minutes outside the MCR.

Automatic actuation of control systems such as Pressurizer Pressure Control System (PPCS), Pressurizer Level Control System (PLCS), Feedwater Control System (FWCS) and Steam Bypass Control System (SBCS) will be credited if it is not affected by the initiating events and has a negligible effect on result of event [6, 7].

SPACE computer code [8] will be used to simulate Nuclear Steam Supply System (NSSS) thermal-hydraulic transient behavior. The SPACE code is a transient thermal-hydraulic analysis code designed for use in best-estimate evaluation of light water reactor systems.

2.2 Acceptance Criteria for DEC-A

The acceptance criteria of DEC-A are defined for physical barriers, including the fuel, Reactor Coolant System (RCS) and containment. Acceptance criteria are satisfied, if Peak Cladding Temperature (PCT) remains below 1,204°C (2,200°F). As for the RCS, the maximum allowable pressure is set to 125% of the design pressure. The allowable containment pressure is set to the Factored Load Category (FLC). Also, the radiological dose limit is set to 10 mSv.

3. Strategy for DEC-A Safety Analysis

3.1 Scenarios and Safety Features for DEC-A

For each DEC-A event, preliminary scenarios and associated safety features during the first 30 minutes following event initiation with no operation action will be described in the respective Sections of the DKN 5&6 PSAR Chapter 20 as follows.

3.1.1 ATWS due to Mechanical Blocking of Rods

In PSAR Section 20.1.1, ATWS due to mechanical blocking of rods is a PIE where the reactor is not scrammed after an AOO due to the mechanical blocking of Control Element Assemblies (CEAs). As the reactor fails to trip by mechanical failure of the CEAs, primary system pressure rises rapidly and Steam Generator (SG) inventories are depleted. As the SG level reaches Low Steam Generator Level (LSGL) setpoints, Passive Auxiliary Feedwater System (PAFS) is actuated simultaneously with the closing of the Main Steam Isolation Valves (MSIVs) and Main Feedwater Isolation Valves (MFIVs) closer. Pilot Operated Safety Relief Valve (POSRV) flow increases rapidly due to increased RCS pressure, and the Emergency Boration System (EBS) is actuated by Emergency Boration Actuation Signal (EBAS).

3.1.2 ATWS due to Failure of RPS

In PSAR Section 20.1.2, ATWS due to failure of RPS is a PIE where the reactor is not scrammed after an AOO due to failure of the RPS. But, Diverse Protection System (DPS) is designed to provide reactor trip functions and passive auxiliary feedwater supply functions.

3.1.3 SBO

In PSAR Section 20.1.3, SBO involves the Loss of Offsite Power (LOOP) and the failure of the onsite Emergency Diesel Generators (EDG). But, the reactor trip can occur by RPS with In-core Protection System (ICOPS) low Reactor Coolant Pump (RCP) shaft speed signal or CEAs drop by interruption to the CEA holding coils due to loss of electric power. Automatic actuation of PAFS removes decay heat from the core.

Alternate AC Diesel Generator (AAC-DG) can mitigate an SBO, which provides backup power to all the equipment necessary for the loads required in the event of an SBO. Also, The Auxiliary Charging Pump (ACP) performs a function to RCS makeup.

3.1.4 TLOFW

In PSAR Section 20.1.4, Total Loss of Feedwater (TLOFW) to the SGs assumes the complete loss of feedwater including the loss of PAFS. TLOFW results in decreasing water level and increasing pressure in the SGs. So, the reactor trip can occur by RPS with LSGL or High Pressurizer Pressure (HPP) signal. Once the operator initiates feed (Safety Injection) and bleed (POSRV) operation, the RCS pressures decreases below safety injection shutoff head, which enables the RCS makeup.

3.1.5 LOCA with Loss of Safety Injection

In PSAR Section 20.1.5, Loss of Coolant Accident (LOCA) with loss of safety injection assumes the loss of all active safety injection after the occurrence of LOCA. When a LOCA occurs, the reactor trip can occur by RPS with High Containment Pressure (HCP) or Low Pressurizer Pressure (LPP) signal. The RCS inventory is challenging to deplete and the core may become uncovered, leading to fuel heat-up. However, the Safety Injection Tanks (SITs) are available since these are passive components. Automatic actuation of PAFS removes decay heat from the core. Since this is a break event, satisfaction with radiological dose acceptance criteria will be evaluated.

3.1.6 Uncontrolled Boron Dilution

In PSAR Section 20.1.6, accident sequences (e.g., heterogeneous boron dilution) that have the potential to cause a large or early release will be practically eliminated according to EUR requirements [2].

3.1.7 LOUHS and Total Loss of Cooling Chain during Normal Operation

In PSAR Section 20.1.7 and 20.1.8, since both events exhibit similar behaviors after event initiation, the loss of all related cooling chains is conservatively assumed. Due to loss of the RCP, Centrifugal Charging Pump (CCP) and Condenser Vacuum (LOCV) is occurred. The reactor trip can occur by RPS with ICOPS low RCP shaft speed signal or HPP signal. Automatic actuation of PAFS removes decay heat from the core. Also, ACP performs a function to RCS makeup.

3.1.8 LOSFPC during Normal Operation

In PSAR Section 20.1.9, the Loss of Spent Fuel Pool Cooling accident (LOSFPC) assumes the failure of both trains of Spent Fuel Pool (SFP) cooling system. The objective of the analysis is to demonstrate that the SFP cooling water can be sufficiently cooled to prevent boiling.

3.1.9 MSGTR

In PSAR Section 20.1.10, Multiple Steam Generator Tube Rupture (MSGTR) results in a direct leakage of primary coolant to the atmosphere with challenge to the bypass of containment. The reactor trip can occur by RPS with ICOPS hot leg saturation margin, LPP or High Steam Generator Level (HSGL) signal. The SGs are isolated by HSGL signal of the affected SG.

The Steam Generator Emergency Blowdown System (SGEBDS) is used to prevent the overfilling of the affected SG. On the other hand, RCS is cooled down by PAFS of the unaffected SG. Also, the safety injection is used to maintain adequate core cooling.

3.1.10 MSLB with Consequential SGTR

In PSAR Section 20.1.11, Main Steam Line Break (MSLB) with consequential Steam Generator Tube Ruptures (SGTR) assumes the concurrent occurrence of a MSLB and a consequential rupture of SG tubes. The primary-to-secondary leak makes the pressurizer pressure and level gradually decrease. Therefore pressurizer pressure reaches safety injection setpoint and safety injection is used to maintain adequate core cooling. Automatic actuation of PAFS removes decay heat from the core. Since this is a break event, satisfaction with radiological dose acceptance criteria will be evaluated.

3.1.11 LORHR

In PSAR Section 20.1.12, Loss of Residual Heat Removal System (LORHR) event is assumed to occur due to the complete loss of shutdown cooling function during mid-loop operation. If the shutdown cooling function is not restored to operate, the core may be uncovered due to the boiling in the core and the steam and/or liquid discharge into the containment. But, the safety injection is used to maintain adequate core cooling.

3.1.12 ISLOCA

In PSAR Section 20.1.13, the design features to mitigate ISLOCA challenges will be evaluated. The system retains its structural integrity throughout the event. Also, any leakage caused by the event is limited to the make-up system capabilities, and offsite doses are limited.

3.2 Dedicated Features for DEC-A

Table 2 describes the dedicated features for DEC-A events, which will be applied to support the DKN5&6 PSAR based on the APR1000 standard design.

Table 2. Dedicated Features for DEC-A

PSAR Section	Dedicated Features
20.1.1	Emergency Boration System (EBS)
20.1.2	Diverse Protection System (DPS)
20.1.3	Alternate AC Diesel Generator (AAC-DG)
20.1.7 20.1.8 20.1.9 20.1.12	Diverse Essential Service Water System (DESWS)
	Diverse Component Cooling Water System (DCCWS)
	Diverse Containment Spray System (DCSS)
	Diverse Ultimate Heat Sink (DUHS)
	Diverse Spent Fuel Pool Cooling System (DSFPCS)
20.1.10	Steam Generator Emergency Blowdown System (SGEBDS)

3.3 General Strategy for DEC-A

In general, decay heat will be removed via PAFS, and the RCS will be depressurized to Shutdown Cooling System (SCS) entry condition by operator action after 30 minutes following event initiation.

According to EUR requirements [2], the design should preferably ensure that the safe state is reached within 24 hours. Therefore, the analysis will be performed to demonstrate that the analysis result reaches a safe state within 24 hours, considering the methodologies and strategies described previously.

In addition, the plant should be designed to maintain safe state within 24 hours in the event of a DEC-A using only permanent equipment. Onsite light Non-Permanent Equipment (NPE) can be credited after 24 hours, onsite heavy NPE after 72 hours and offsite NPE after 7 days.

Additionally, DKN 5&6 has diverse systems as described in Section 3.2, such as the Diverse Essential Service Water System (DESWS), Diverse Component Cooling Water System (DCCWS), and Diverse Containment Spray System (DCSS), which provide heat removal to the Diverse Ultimate Heat Sink (DUHS). The Diverse Spent Fuel Pool Cooling System (DSFPCS) can also be activated to remove decay heat via the DCCWS. These diverse systems enable the plant to be maintained in a safe state through operator action.

4. Preliminary Safety Analysis Results for DEC-A

This section presents the preliminary safety analysis results using SPACE code based on best-estimate methodologies and strategies as described in Section 2.1 for ATWS due to mechanical blocking of rods (PSAR Section 20.1.1), one of the DEC-A events. The nodalization scheme for this analysis using the SPACE code is shown in Figure 1.

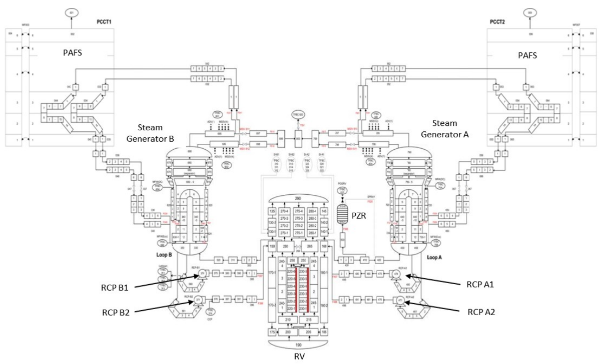

Figure 1. SPACE nodalization in APR1000

The most critical challenge of ATWS due to mechanical blocking of rod is a threat to the integrity of the RCS pressure boundary due to overpressurization. ATWS with Loss of Normal Feedwater (ATWS-LONF) is selected as one of the limiting event in terms of RCS overpressurization.

The following assumptions are applied to the analysis of the ATWS-LONF event. Due to the assumed loss of feedwater flow following a Loss of Normal Feedwater (LONF) initiating event, the heat transfer rate from the RCS to the secondary system decreases significantly. This reduction in heat removal causes the RCS temperature to rise; the resulting thermal expansion of the primary coolant drives an increase in both pressurizer level and pressure. Although reactor trip signal is generated by LSGL setpoint as SG inventory depletes, the assumed ATWS condition (caused by the mechanical blocking of rods) prevents a reactor scram insertion. Consequently, the reactor relies solely on the negative reactivity feedback of the Moderator Temperature Coefficient (MTC) to reduce core power.

Also, the power level eventually stabilizes at a point that matches the heat removal capacity of the PAFS. Throughout the transient, RCS pressure is mitigated by the POSRV. However, the peak RCS pressure occurs following a solid pressurizer condition (PRZ full water level), where the lack of a steam cushion leads to a rapid pressure spike.

From the perspective of nuclear design data, the negative MTC is an inherent safety feature that provides negative reactivity feedback as coolant temperature increases, thereby reducing core power and mitigating the rise in RCS pressure. Consequently, peak RCS pressure is observed when the MTC is least negative, which typically corresponds to higher core power conditions. For this analysis, the Moderator Density Coefficient (MDC) value corresponding to the least negative MTC of the first fuel cycle is assumed, representing the bounding case for RCS overpressurization.

Figures 2 to 8 show the transient event sequences. As described above, core power (Fig. 2) is affected by reactivity feedback from changes in moderator and fuel temperatures. SG inventory depletion reduces heat transfer, resulting in increased RCS temperature (Fig. 3). This leads to primary coolant expansion, increasing RCS pressure (Fig. 4) and pressurizer level (Fig. 5). As RCS pressure increases, POSRVs open, resulting in POSRV discharge flow (Fig. 6). As SG level decreases, PAFS is actuated (Fig. 7). Core reactivity is maintained negative through continuous boron injection by EBS (Fig. 8), counteracting positive reactivity insertion due to RCS cooldown.

Peak RCS pressure occurs at approximately 106.4 sec (Fig. 4), maintaining a large margin to the acceptance criteria (125% of the design pressure). In addition, it is confirmed that the radiological dose, based on the analysis results, is below the acceptance criteria (10 mSv).

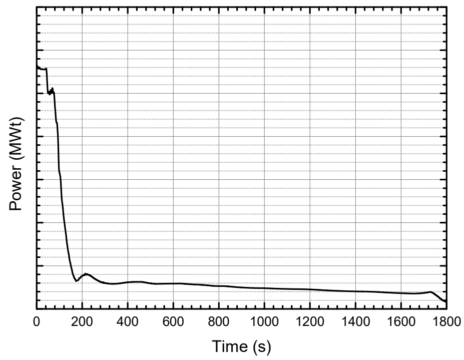

Figure 2. Core Power (MWt)

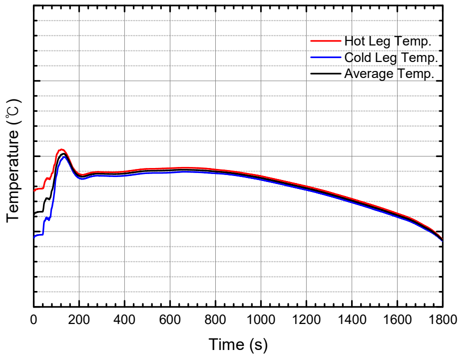

Figure 3. RCS Temperature (℃)

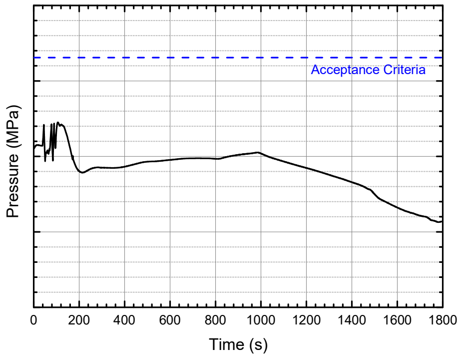

Figure 4. RCS Pressure (MPa)

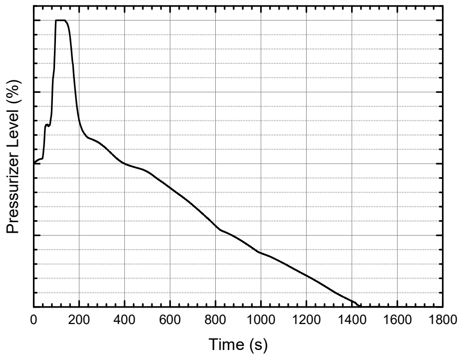

Figure 5. Pressurizer Level (%)

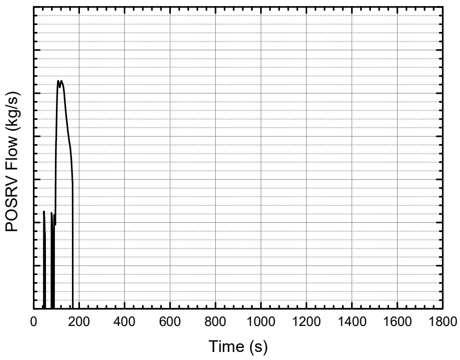

Figure 6. POSRV Flow Rate (kg/s)

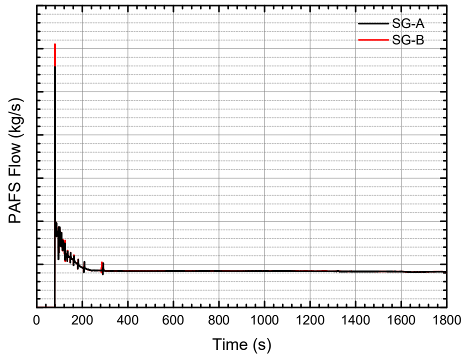

Figure 7. PAFS Flow Rate (kg/s)

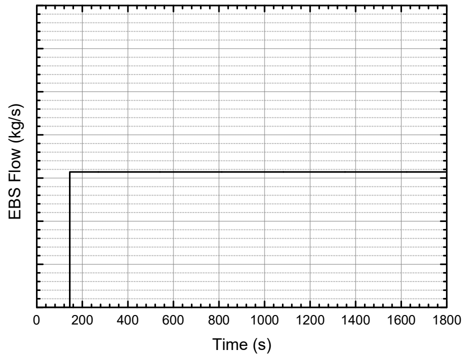

Figure 8. EBS Flow Rate (kg/s)

Figures 2 to 8 show the analysis results with no operator action for the first 30 minutes. After 30 minutes, the operator actuates to depressurize the RCS by Reactor Coolant Gas Vent System (RCGVS), and the RCS temperature continues to be cooled by PAFS. Figures 9 to 10 shows that RCS pressure and temperature reach the SCS entry conditions before about 6 hours respectively.

Consequently, the analysis is performed to demonstrate that the analysis result reaches a safe state within 24 hours.

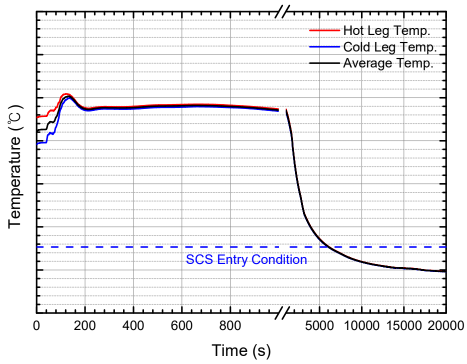

Figure 9. Temperature (℃) – Long Term

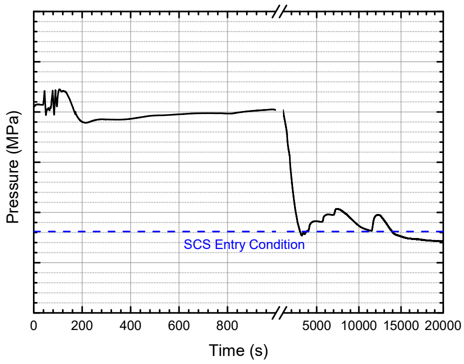

Figure 10. RCS Pressure (MPa) – Long Term

EBS is a dedicated feature for mitigating ATWS due to mechanical blocking of rods event as described in Section 3.2. Therefore, the EBS performance including Emergency Boration Tank (EBT) capacity, boron concentration, and EBS pump flow rate is evaluated based on the Safety Injection Pump (SIP) shutoff head. Once SIP injection begins, the operator is able to maintain the plant in a safe state. Additionally, EBS actuation signal referred to as EBAS is determined by combining signals considering for ATWS conditions caused by the mechanical blocking of rods.

Following EBAS generation, successful EBS (Fig. 8) operation maintains core subcriticality by injecting borated water into the RCS until RCS pressure decreases below the safety injection shutoff head. Figure 11 shows that the total reactivity is confirmed to remain negative following boron injection by EBS.

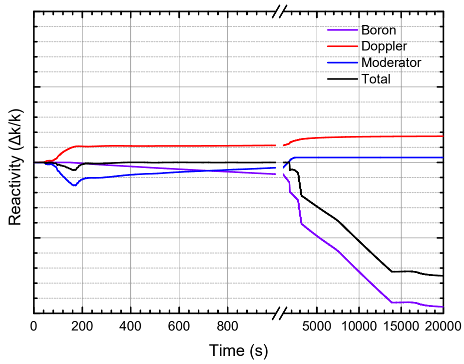

Figure 11. Reactivity (Δk/k) – Long Term

5. Conclusions

This paper presents the definition and classification of DEC-A events. Based on this assessment, 13 DEC-A events were identified, and corresponding analysis methodologies and strategies were established.

Methodologies and strategies were provided for DEC-A analyses, including assumptions, acceptance criteria, event scenarios, safety and dedicated features, and mitigation strategies.

ATWS due to mechanical blocking of rods, one of the DEC-A events, was analyzed using SPACE code with the presented methodologies and strategies. The preliminary analysis results demonstrated that acceptance criteria were satisfied and a safe shutdown state was reached within 24 hours. Additionally, performance evaluations of dedicated features (e.g., EBS) were performed.

As the DKN5&6 final design process continues, each DEC-A events will be re-analyzed to support Chapter 20 of the DKN5&6 PSAR. This process will be improved following additional evaluations such as cliff-edge effect.

Furthermore, the DEC-A methodologies and strategies can be refined through clarification processes to ensure compliance with Czech and additional regulatory requirements.

ACKNOWLEDGEMENTS

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