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Deterministic and Probabilistic Analysis against Anticipated Transient Without Scram

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1. Introduction

An Anticipated Transient Without Scram (ATWS) is an Anticipated Operational Occurrences (AOOs) accompanied by a failure of the reactor trip when required. By a suitable combination of inherent characteristics and diverse systems, the reactor design needs to reduce the probability of the ATWS and to limit any Core Damage and prevent loss of integrity of the reactor coolant pressure boundary if it happens.

In EU-APR, which is a modified design of the APR1400 to penetrate the European nuclear market, diverse safety systems to prevent and mitigate the ATWS have been implemented. This study focuses on the deterministic analysis for the ATWS events with respect to Reactor Coolant System (RCS) over-pressure and fuel integrity for the EU-APR. Additionally, this report presents the Probabilistic Safety Assessment (PSA) reflecting those diverse systems.

2. Development of Diverse Safety System

2.1 Phenomenological Sequence of ATWS

ATWS events which cause plant condition excursions resulting in close to or over the acceptance criteria involve a mismatch of power produced by the reactor core and power removed from the RCS. The mismatch may be initiated either by an unexpected increase in reactor power or an unexpected decrease in heat removal from the RCS. In either case, failure of the reactor scram causes the mismatch between heat generation and removal in the RCS.

Unexpected reactor power increases can be caused by certain failures in the reactivity control system and changes in soluble poison concentration. Unexpected decreases in RCS heat removal can be caused by various disturbances including reduction (or elimination) of main steam flow, reduction or termination of feedwater flow to the Steam Generator (SG), reduction of reactor coolant flow, or changes in feedwater temperature.

All ATWS events resulting in excessive core power production over the rate of heat removal from the RCS cause the increase in RCS pressure. Over-pressurization of the RCS is caused by expansion of the reactor coolant as its temperature increases. A major objective of ATWS analysis is to evaluate the amount of RCS pressure rise and its consequences, particularly with regard to RCS pressure boundary integrity.

Another consequence of the mismatch of reactor core power and RCS heat removal is the increase of stored energy within the reactor fuel and increased potential for fuel cladding degradation. Increased stored energy, and associated increased fuel temperature, can occur directly because of inability of the fuel to rapidly conduct energy to its surface or indirectly because of the degradation of fuel surface heat transfer caused by perturbations to the reactor coolant. Other potential failure mechanisms include clad overheating due to Departure from Nucleate Boiling (DNB) and cladding oxidation at high temperature.

2.2 Scope

This report primarily covers the physical phenomena which characterize ATWS in the EU-APR and the principal analytical considerations necessary to mitigate ATWS consequences. This report also presents the analysis results of the following ATWS events for the EU-APR. The seven ATWS events are chosen among AOOs and listed as follows:

- Inadvertent withdrawal of control rod bank
- Inadvertent boric acid dilution
- Excess increase of steam flow
- Spurious opening of a steam generator safety valve or other secondary side depressurization caused by a single failure
- Partial loss of core coolant flow
- Loss of main feedwater flow to steam generators
- Total loss of off-site power (< 2 hours), assuming house load operation failure

The first four events listed above result in core power increase from the initial value. The next three ATWS events result in the decrease in heat removal from the RCS. In this report, the following four events out of seven are selected based on the experience for the reference plant due to the major concerns derived from the consequences of the expected high primary system pressure and challenge to the fuel integrity.

- ATWS 1: Inadvertent withdrawal of control rod bank (Hot zero power level and 100% power level)
- ATWS 2: Excess Increase of Steam Flow
- ATWS 3: Loss of main feedwater flow to steam generators
- ATWS 4: Loss of off-site power

2.3 Identification of Events and Causes

An event of excess increase of steam flow is defined as any rapid increase in steam generator steam flow, other than a steam line rupture. An increase in steam flow may be caused by the inadvertent opening of a turbine bypass valve or main steam atmospheric dump valve.

The Loss of Normal Feedwater Flow (LONF) event may be initiated by losing two or more of the three operating main feedwater pumps or by a spurious signal being generated by the feedwater control system resulting in a closure of the feedwater control valve.

For the case of the real power plant, the Unit can be shift to the house load operation. For the safety analysis, however, the Loss of Offsite Power (LOOP) event is conservatively assumed as a complete loss of AC electrical power. As a result, electrical power would be unavailable for the reactor coolant pumps, the steam generator feedwater pumps, and the main circulating water pumps, and for maintaining the condenser vacuum. Therefore, the Emergency Diesel Generators (EDGs) are only operable during the LOOP event to provide electrical power for safety-related components.

A CEA bank Withdrawal (CEAW) event is defined as a failure in the Control Element Drive Mechanism (CEDM), Control Element Drive Mechanism Control System (CEDMCS), Reactor Regulating System (RRS), or as a result of operator error. The CEAW event initiates from the withdrawal of lead bank which is maximally inserted as 28% of its length causing a certain rate of positive reactivity insertion. This positive reactivity insertion, however, causes the core power, core average heat flux, and RCS temperature and pressure to rise. Moreover, the minimum DNBR decreases by the positive reactivity insertion. The increase in RCS pressure activates the pressurizer sprays which mitigate the pressure rise.

2.4 Diverse Systems

The ATWS mitigation systems are needed to be automatically initiated during the event and no credit is taken for manual action by operator during the 30 minutes according to EUR [1] and YVL [2].

The diverse reactivity control system to shut down the reactor is needed to consider the failure of scram rod insertion caused by mechanical problem. Emergency Boration System (EBS) is designed for the EU-APR to provide diverse reactivity control functions in the event of ATWS and is available to ensure automatic boration of the RCS and reach the sub-criticality.

Diverse Protection System (DPS) provides all the Reactor Coolant Pumps (RCPs) shut down on the low steam generator level signal and Emergency Boration Actuation Signal (EBAS) as well as automatic reactor trips on high pressurizer pressure or high containment pressure and an automatic actuation of the auxiliary feedwater on low steam generator level. The EBAS are triggered on reactor trip signal of DPS and no bottom signal of dropped rods.

3. Deterministic Safety Analysis

3.1 Analysis Methodology

Input parameters for Pilot-Operated Safety and Relief Valve (POSRV), Main Steam Atmospheric Dump Valve (MSADV) and Auxiliary Feedwater System (AFWS) are shown in Table 1.

Table 1: Major Parameters for the ATWS Analyses

Parameters	Value
POSRV Rated Flow Rate Per One Valve [kg/s]	135.0
MSADV Rated Flow Rate Per One Valve [kg/s]	252.0
Auxiliary Feedwater Flow Rate per One SG [m^3/hr]	124.9
Auxiliary Feedwater Actuation Setpoint [% of Wide Range]	25.0

Tables 2 and 3 summarize significant parameters for the analyses of events 1 through 4, respectively.

Table 2: Significant Parameters for ATWS 1

Parameters	Value
EBS Rated Flow Rate Per One Pump [m^3/hr]	11.4
CEA Bank Withdrawn Length [m]	1.067
Speed of Withdrawing CEA Bank Motion [m/min]	0.762
Maximum Reactivity Insertion Rate by CEA Bank Withdrawal at 100% Power Level [10^{-4} $\Delta\rho$/m]	24.80*/24.41**
Maximum Reactivity Insertion Rate by CEA Bank Withdrawal at Hot Zero Power Level [10^{-4} $\Delta\rho$/m]	90.16**
Moderator Temperature Coefficient (FTC) [10^{-4} $\Delta\rho/^{o}C$]	-0.371* /-1.6675**
Fuel Temperature Coefficient (FTC) [10^{-4} $\Delta\rho/^{o}C$]	-0.0734* /-0.0869**

* in case of over-pressure and minimum DNBR
** in case of sub-criticality

Table 3: Significant Parameters for ATWS 2, 3 and 4

Parameters	Value
Moderator Temperature Coefficient (FTC) [10^{-4} $\Delta\rho/^{o}C$]	-0.354* /-1.6675**
Fuel Temperature Coefficient (FTC) [10^{-4} $\Delta\rho/^{o}C$]	-0.0678* /-0.0869**
Critical Flow Model for POSRVs (only needed for Event 3)	Homogeneous Equilibrium Model (HEM)

* in case of over-pressure and minimum DNBR
** in case of sub-criticality

For ATWS analysis, best estimate methodology is used and a single failure criterion is applied in accordance with YVL [2].

Two digital computer programs are used in the quantitative evaluation for ATWS events, RETRAN-3D and CETOP-D. RETRAN-3D code calculates NSSS thermal-hydraulic responses to the initiating events for a wide range of operating condition. CETOP code calculates the minimum value for DNBR which serves as a measure for the core thermal margin.

3.2 Acceptance Criteria

Acceptance criteria for ATWS events in YVL [2] are as follows:

- Core coolable geometry retained
- Cladding temperature < 1,200℃
- Pressures in the primary and secondary circuit < 120% of the design pressure

To determine the fuel integrity, the value of 1.29 is applied for the minimum DNBR because of Specified Acceptable Fuel Design Limit (SAFDL) for PLUS7.

Radiological consequences are not of concern since the physical barriers are maintained during the accident when aforementioned acceptance criteria are met.

3.3 Results

Four events were quantitatively analyzed in terms of RCS pressure, minimum DNBR and reactivity during the events. It was found that fuel cladding integrity was maintained and sub-criticality was achieved with sufficient margins for each event within 30 minutes, owing to the inherent characteristics and automatic diverse systems. The analysis results are summarized in Table 4.

In these analyses, ATWS 1 was most challenging to the RCS integrity. The maximum RCS pressure at the RCP discharge is 20.07 MPa which is below acceptance criteria of 20.68 MPa (120% of design pressure), ensuring primary system integrity, as shown in Figure 1. The maximum steam generator pressure is 8.195 MPa which is below acceptance criteria of 9.92 MPa (120% of design pressure), ensuring secondary system integrity. The minimum DNBR is 2.023 which remain above 1.29 during the transient, as presented in Figure 2. Therefore, there is no fuel cladding failure. As shown in Figure 3 and 4, the total reactivity of the core maintains sub-criticality after 119 seconds for 100% power level and 879 seconds for hot zero power level, respectively.

Table 4 Results of ATWS Analyses

	ATWS 1	ATWS 2	ATWS 3	ATWS 4
Maximum RCS pressure [% of the design pressure]	116	106	107	109
Minimum DNBR	2.023	1.740	2.026	1.913
Time to Sub-criticality [sec]	119*/ 879**	116	960.6	337

* 100% power level
** hot zero power level

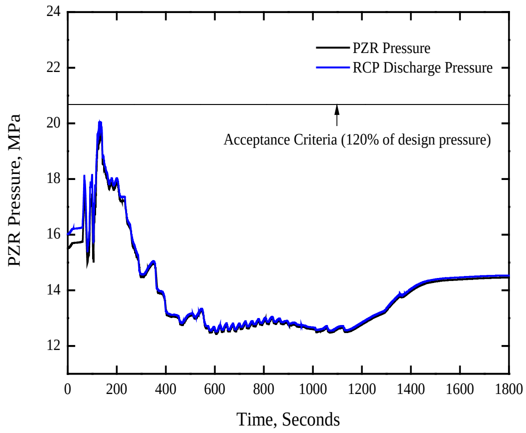

Figure 1: Pressurizer Pressure during ATWS 1 (Over-pressure Aspect)

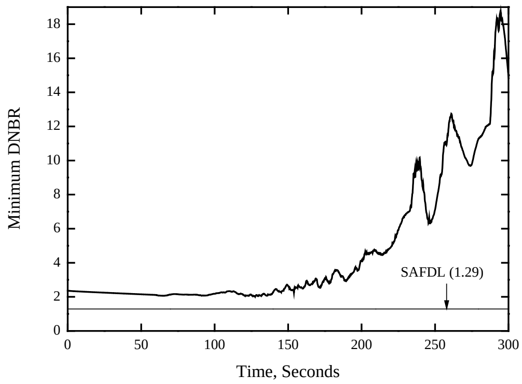

Figure 2: Minimum DNBR during ATWS 1 (100% Power Level, Re-criticality Aspect)

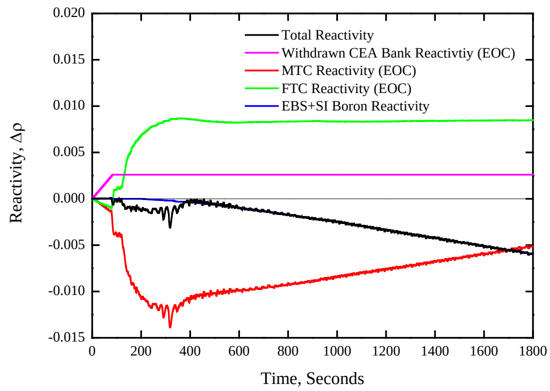

Figure 3: Reactivity during ATWS 1 (100% Power Level, Re-criticality Aspect)

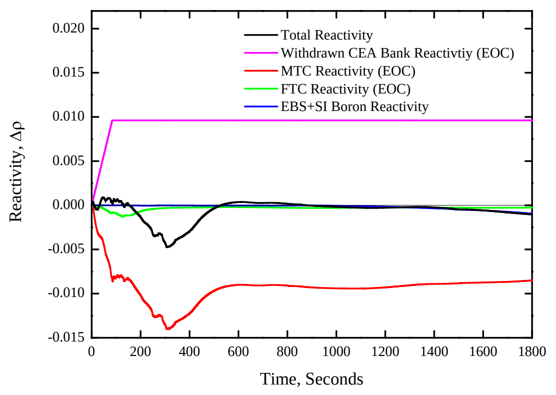

Figure 4: Reactivity during ATWS 1 (Hot Zero Power Level, Re-criticality Aspect)

4. Probabilistic Safety Assessment

4.1 Analysis Methodology

SAREX is used to develop the event tree models [3]. The SAREX is integrated PSA software that provides the ability to create and evaluate fault trees and event trees. The event tree models develop for each initiating event. The results of the accident sequence analysis are the identification of the individual core damage sequences, and the analysis requirements for determining the timing and progression of each accident sequence. The timing information is required in order to evaluate the impact of the operator actions, and the time of occurrence of the automatic systems initiation signals.

For each initiating event, progression of potential scenarios leading to either a safe state or to core damage is modeled using an event tree. Functions required for mitigating the accident to prevent core damage are included across the top of the event tree.

Following generic sources are reviewed to develop the initiating event list for EU-APR PSA Level 1 internal events;

- NUREG/CR-5750 [4]
- NUREG/CR-3862 [5]
- EPRI NP-2230 [6]

The frequency of ATWS is estimated by using the plant specific fault trees generated in accident sequence quantification process in which a failure of reactor scram is combined with initiating events during accident sequence quantification process.

4.2 Result

ATWS is defined as the occurrence of an anticipated transient with a failure to scram due to the failure of the control rods to insert caused by mechanical failure/binding or the failure of the Reactor Protection System (RPS) and the DPS to generate the trip signal.

The generic events modeled in the ATWS event tree in are described below.

- Adverse Moderator Temperature Coefficient
- POSRV Failure to Open
- POSRV fails Reclose
- Consequential Steam Generator Tube Rupture due to the pressure excursion following an ATWS
- Secondary Heat Removal Failure
- Failure of Delivering Boron via Charging Pump and EBS

For event tree analysis, seven accident sequences were derived from the combination of these generic events. Excluding one event with very low frequency, six accident sequences could be derived. Consequently, the individual core damage sequences regarding the ATWS were identified by SAREX code.

5. Conclusions

The analysis performed for the ATWS event indicates that the NSSS could be reached to controlled and safe state due to the addition of boron into the core via the EBS pump flow upon the EBAS by DPS. Decay heat is removed through MSADVs and the auxiliary feedwater. During the ATWS event, RCS pressure boundary is maintained by the operation of primary and secondary safety valves. Consequently, the acceptance criteria were satisfied by installing DPS and EBS in addition to the inherent safety characteristics.

The result of PSA level 1 internal events showed that the basic aim of EUR [1] probabilistic safety objective in terms of Core Damage Frequency (CDF) was met. It was found that the contribution of ATWS was less than 10% of the total CDF applying aforementioned diverse design features.

REFERENCES

[1] “European Utility Requirements for LWR Nuclear Power Plants," Revision D, October 2012, Volume 2

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[5] NUREG/CR-3862, Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment, May 1985.

[6] EPRI NP-2230, “ATWS: A Re-Appraisal, Part 3: Frequency of Anticipated Transients,” Interim Report, Electric Power Research Institute, January 1982