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An ATWS Analysis for EU-APR1400 Following the European Utility Requirement

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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. This paper presents the results of the evaluation of the ATWS events with respect to Reactor Coolant System (RCS) overpressure and re-criticality for the European APR1400 (EU-APR1400) according to European Utility Requirement (EUR) [1].

2. Methods and Results

According to the EUR, two methods shall be applied to the safety analyses which are the conservative analysis method supplemented with sensitivity analyses, and the best estimated method supplemented with uncertainty analysis. ATWS is classified as a Design Extension Condition (DEC) event, the best estimated method supplemented with uncertainty analysis shall be used in the analysis. Therefore, in this paper, realistic best estimate assumptions are used.

2.1 Selection of Transients

In this paper, to evaluate the capability of EU-APR1400 design against ATWS, the following four cases are quantitatively analyzed to assess their impact on the RCS overpressure protection and return to criticality. These transients are selected based on the experience for APR1400 and provide sufficient characterization of the response of the APR1400 design to ATWS.

- Inadvertent withdrawal of control rod bank
- Excess increase of steam flow
- Loss of main feedwater flow to steam generator
- Loss of off-site power

2.2 Computer Program

Nuclear Steam Supply System (NSSS) thermal hydraulic responses to the limiting events with respect to RCS overpressure and return to criticality for the EU-APR1400 are simulated using the RETRAN-3D [2]. Figure 1 shows the RETRAN-3D nodalization of the primary and secondary systems and the major components of the EU-APR1400.

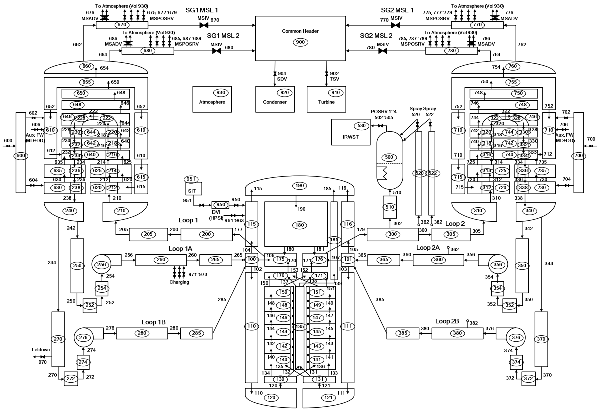

Fig. 1 RETRAN 3-D Nodalization for EU-APR1400

2.3 Initial Conditions and Assumptions

Nominal initial conditions and design data are used in the analysis. The major initial conditions are summarized in Table 1. And the functions and related systems needed in ATWS events up to the controlled state and 30 minutes after event initiation are summarized in Table 2.

Table 1. Initial Conditions for ATWS events

Parameters	Initial Values
Core Power, MWt	3,983
PZR Pressure, MPa	15.51
Core Inlet Temperature, °C	291.2
RCP Flow, kg/sec/pump	5,238.5
PZR Liquid Inventory, m^3	34.6
SG Pressure, MPa	7.584~6.895*

* 0% Power : 7.584 MPa, 100% Power : 6.895 MPa

Table 2. Functions Used in ATWS events and Assumptions

Functions and Systems		ATWS Assumption
Functions	System/Component	
Control of PZR Press.	PZR spray	O
Control of PZR level	PLCS	X
Core Power Reduction	RPCS	X
Control of SGs Press.	SBCS	X
Control of SG level	FWCS	X
Diverse Protection	DPS	O
Overpressure Protection	MSADV	O
	MSPOSRV	O
	POSRV	O
Borated Water Injection	SIS	O
	EBS	O
Cooldown by SGs	AFWS	O
SGs Isolation	MSIS	O

2.4 Single Failure

The most adverse two Single Failures (SFs) which are failure of one Emergency Boration System (EBS) pump and one POSRV failure are additionally considered to conservatively simulate the ATWS in terms of RCS pressurization and re-criticality, respectively.

2.5 Reactor Physics Data

The range of best estimate value for Moderator Temperature Coefficient (MTC) is -0.354 ~ -1.6675 (10^{-4} $\Delta\rho/^{o}C$), which covers the whole fuel cycle. And the range of best estimate value for Fuel Temperature Coefficient (FTC) is -0.0678 ~ -0.0869 (10^{-4} $\Delta\rho/^{o}C$), which covers the whole fuel cycle. MTC and FTC are assumed the value of least negative with respect to overpressure. In case of sub-criticality, MTC and FTC are conservatively assumed the value of most negative.

2.6 Analysis Results

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. In general, 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 makes the mismatch between heat generation and removal in the RCS. 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, 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. Overpressurization of the RCS is caused by expansion of the reactor coolant as its temperature increases. The ATWS mitigation systems such as Engineering Safety Features Actuation Signal (ESFAS), EBS and Diverse Protection System (DPS) are automatically initiated during the event.

As shown in Figure 2 and Figure 3, the maximum primary pressure remains below 120% of design pressure (20.09 MPa) and the total reactivity of the core maintains sub-criticality in the core mitigation for all ATWS events with respect to overpressure and re-criticality, respectively. The time of reaching sub-criticality for each event is presented in Table 3.

Table 3. Time of Reaching Sub-Criticality

ATWS events	Time (Seconds)
IOMSADV	116
LONF	960.6
LOOP	3,375
CEAW	119

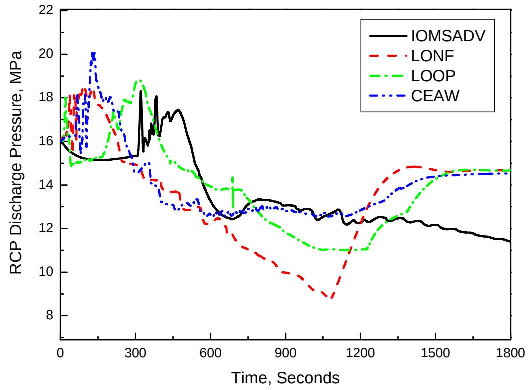

Fig. 2 RCS Pressure vs. Time (100% Power level)

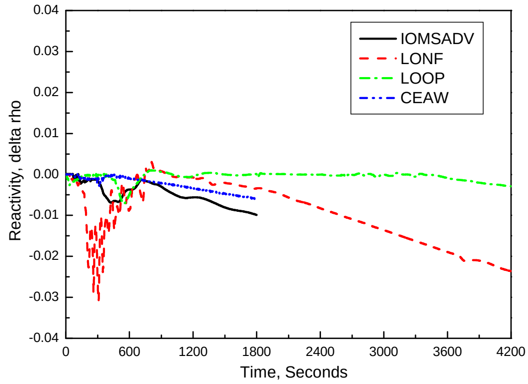

Fig. 3 Reactivity vs. Time (100% Power level)

3. Conclusions

This paper evaluates the ATWS impact on the EU-APR1400 by applying EUR. Based on the results of safety analysis for ATWS events, all the acceptance criteria for EUR can be satisfied due to the proper functioning of ATWS mitigation systems. However the four events are investigated only in this paper, and hence the results of this paper can not be concluded that the EU-APR1400 design satisfy all requirements for the EUR. Therefore, a further study for all Design Basis Event Category 2 (DBC2) events with ATWS needs to be performed in order to assess the comprehensive impact of ATWS events for the EU-APR1400 design.

REFERENCES

[1] European Utility Requirements for LWR Nuclear Power Plants, Revision D, Volume 2, Chapter 1, p. 44, October 2012.
[2] EPRI NP-7450, “RETRAN 3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,” Computer Code Manual, Computer Simulation & Analysis, Inc. and Electric Power Research Institute, December 1997.