Transactions of the Korean Nuclear Society Autumn Meeting
Yeosu, Korea, October 25-26, 2018

A Study on Fuel Integrity during an Inadvertent Opening of Steam Generator Relief or Safety Valves Event with Delayed Scram

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1. Introduction

Event categorization of design basis condition (DBC) 2 with delayed scram is introduced as DBC 3 for EUR [1]. Delayed scram means the relevant scram (trip) is assumed on the second actuation limit reached during event sequences. The limiting analysis case with delayed scram is determined on screen analysis procedure, and an inadvertent opening of steam generator relief or safety valves with delayed scram is chosen as a limiting event for the fuel integrity. To check the fuel integrity during an inadvertent opening of steam generator relief or safety valves with delayed scram, a computer code of RETRAN-3D is used. Screen analysis procedure to determine the relevant event is described herein, and the results are shown to evaluate fuel integrity.

2. Methods and Results

The screen analysis procedure is established to determine the limiting one among the events with delayed scram. In this section the screen analysis procedure and the thermal hydraulic results are given.

2.1 Screen and Analysis Procedures

The screen and analysis procedure of the limiting event with delayed scram is shown in Fig. 1. At first, the analyst evaluates each event qualitatively based on plant operating condition and boundary conditions, and then selects the limiting event with the delayed scram. Next, the analyst performs the thermal-hydraulic analysis for the limiting event using a computer code such as RETRAN-3D. If additional reactor trip (delayed trip) is required to meet the acceptance criteria for the relevant event, the trip signal shall be added and its setpoint shall be determined.

Events related to fuel degradation among DBC 2 events are as follows;

- Increase in main steam flow
- Inadvertent opening of steam generator relief or safety valves
- Increase in feedwater flow
- Uncontrolled control element assembly withdrawal from subcritical or low power
- Uncontrolled control element assembly withdrawal at full power
- Inadvertent actuation of the pressurizer spray

With respect to fuel integrity, an inadvertent opening of steam generator relief or safety valves event is selected as representative event among DBC 2 events via the screen analysis procedure.

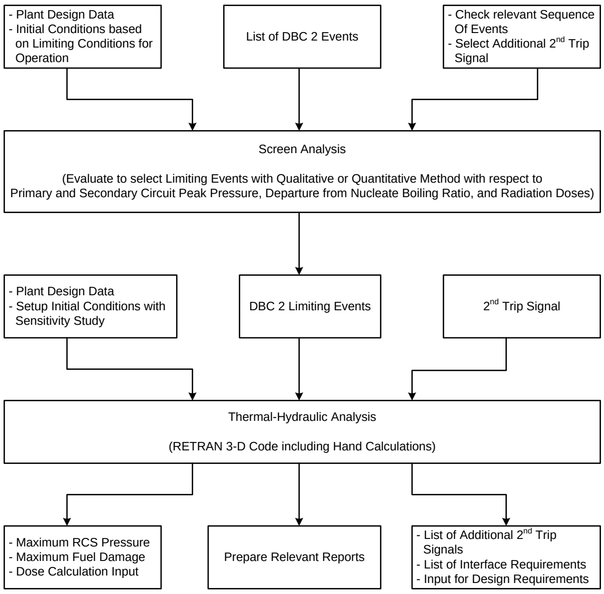

Fig. 1. Screen and Analysis Procedure

2.2 Reactor Trip

Primary and secondary reactor trips for an inadvertent opening of steam generator relief or safety valves event are on variable overpower and manual, respectively. Therefore, for an inadvertent opening of steam generator relief or safety valves with delayed trip event, the secondary trip is determined as a manual trip by the operator.

2.3 Computer Program

The computer program, the RETRAN-3D [2], is used in the quantitative evaluation of the DBC 2 events with delayed scram. Sections 2.3 through 2.6 provide the relevant information.

RETRAN-3D code is a transient thermal-hydraulic analysis code designed for use in best-estimate evaluation of light water reactor systems, and RETRAN-3D nodalization of the primary and secondary systems and the major components of the

EU-APR, which is a modified APR1400 to meet EUR, are used.

RETRAN-3D modeling consists of 137 control volumes including core model (split core) for NSSS major systems and above 190 junctions for connecting each control volume or defining boundary conditions. Trip and control cards are also used to simulate the relevant setpoints and response times of the related reactor protection and control systems.

2.4 Acceptance Criteria

Acceptance criteria on fuel integrity for an inadvertent opening of steam generator relief or safety valves with delayed scram is that the number of fuel rods reaching the heat transfer crisis may not exceed 1% of the total number of fuel rods in the reactor [1].

2.5 Selection and Preparation of the Input Data

The major initial conditions used are summarized in Table 1.

Table 1. Initial Conditions

Parameters	**Nominal Values**
Core Power Level, MWt	3,983
Pressurizer Pressure, MPa	16.03
Core Inlet Temperature, ^{o}C	296.1
Reactor Coolant Pump Flow, kg/sec per pump	5,247.8
Pressurizer Water Level, m^3	39.9
Steam Generator Pressure, MPa	6.895

For an inadvertent opening of steam generator relief or safety valves, the no single failure has an effect on the consequence because the minimum DNB occurs within a few second after manual reactor trip (second trip). This is verified by the previous evaluation of an APR1400 design.

Analysis for the inadvertent opening of a steam generator relief or safety valves is carried out with assumption that all limitation and protection systems belonging to safety classes 2 and 3 work. Non-safety classified systems are credited only when they have detrimental effect on transient management.

The plant is assumed to be operating at rated power (3,983 MWt). The sticking of the most reactive control rod in the reactor scram system is conservatively assumed even though it is not required to be postulated. The MTC and Doppler reactivity are assumed the value of -5.4 x 10-4 Δρ/°C and least negative, the least favorable for the inadvertent opening of a steam generator relief or safety valves, respectively

2.6 Results

The opening of a steam generator relief or safety valves increases the rate of heat removal by the steam generators, causing cooldown of the RCS. Due to the negative moderator temperature coefficient, core power increases from the initial value of 102% of rated core power, reaching a new stabilized value of 113%.

The feedwater control system, which is assumed to be in the automatic mode, supplies feedwater to the steam generators such that steam generator water levels are maintained.

The core power increase and reaches variable overpower reactor trip setpoint (103.5%). But, the reactor trip does not occur on variable overpower trip (1st trip) because the first reactor trip is not credited. And then, the operator manually trips the reactor at 1,800 seconds (2nd trip).

Table 2 shows the relevant sequence of event summarizing the major events.

Table 2. Sequence of Event

Time (sec)	**Event**	**Setpoint or Value**	**Remark**
0.0	Inadvertent Opening of a steam generator relief or safety valves	-	
7.2	Variable Overpower Trip occurs, %P	103.5	1st reactor trip
1,800	Operator Initiates Manual Trip	-	2nd reactor trip
1,800.05	Minimum Transient DNBR	-	1.325
1,800.1	Reactor Trip Breaker open/Turbine Trip	-	
1,976	Pressurizer Pressure reaches Safety Injection Actuation Signal Analysis Setpoint, kg/cm^2A	122	
2,121	Steam Generator Pressure reaches Main Steam Isolation Signal Analysis Setpoint, kg/cm^2A	57.1	
2,127.35	MSIVs close completely	-	
2,132.35	MFIVs close completely	-	
3,000.0	Operator Manually closes a steam generator or safety valve	-	
3,600.0	Operator Initiates Plant Cooldown	-	

Fig. 2 and Fig. 3 show core power and pressurizer pressure, respectively, and Fig. 4 shows secondary pressure, and Fig.5 shows the resultant minimum DNBR, and the minimum DNBR value is 1.325.

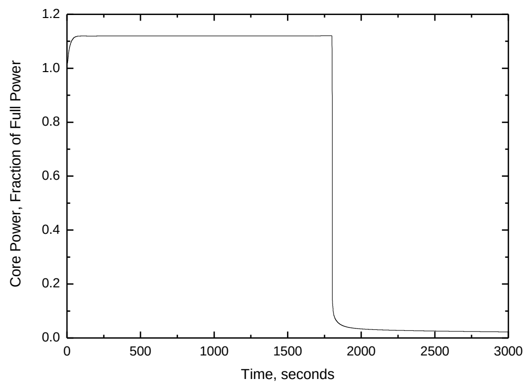

Fig. 2. Core Power vs. Time

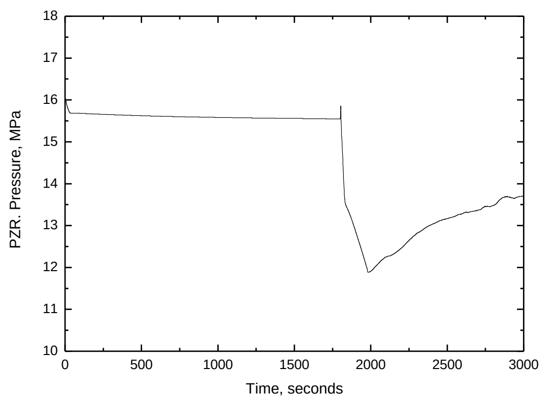

Fig. 3. Pressurizer Pressure vs. Time

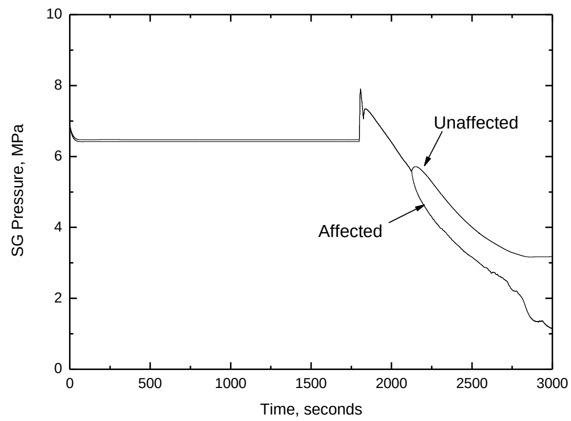

Fig. 4. Steam Generator Pressure vs. Time

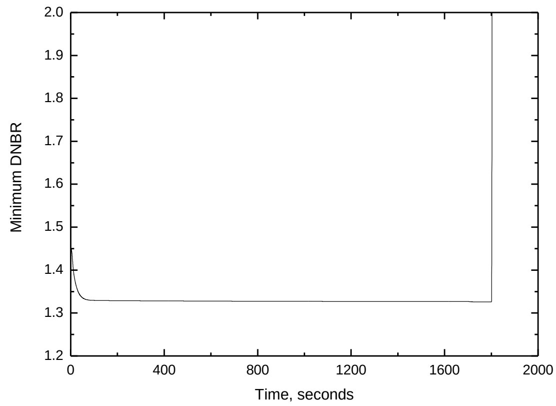

Fig. 5. Minimum DNBR vs. Time

3. Conclusions

For an inadvertent opening of steam generator relief or safety valves with delayed scram, the acceptance criteria for fuel integrity is that the number of fuel rods reaching the heat transfer crisis may not exceed 1% of the total number of fuel rods in the reactor. However, the transient DNBR decreases, reaching a minimum value of 1.325 which is above the specified acceptable fuel design limit (SAFDL) value of 1.29, ensuring the fuel integrities.

REFERENCES

[1] European Utility Requirements for LWR Nuclear Power Plant, Revision D, 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.