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An Evaluation of Non-LOCA Events with Concurrent Common Mode Failure in Digital Plant Protection System for KNGR

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ABSTRACT

An evaluation has been performed qualitatively and quantitatively to determine the intrinsic capability of the Korean Next Generation Reactor (KNGR) design in coping with non-LOCA transients with concurrent Common Mode Failure (CMF) in the digital Plant Protection System (PPS). A best-estimate analysis methodology has been developed and utilized since design bases events with concurrent CMF in digital PPS are categorized as beyond design bases events. Due to diverse means not affected by CMF and a sufficient available over-power margin in KNGR design, the event consequences are well within the acceptance criteria for the events with CMF. In addition, the KNGR design offers sufficient safety margin against non-LOCA events without operator actions up to 30 minutes after the initiation of an event even with CMF.

I. INTRODUCTION

The digital Plant Protection System (PPS) to be designed into the KNGR could be vulnerable to CMF caused by software error, which could defeat the redundancy configured in the hardware architecture. The regulatory policy on CMF in the protection system software specifies that the licensee perform a systematic evaluation which shows the Defense-in-Depth and Diversity (D-in-D&D) capability of the plant design to cope with the design bases events accompanied by CMF in digital based PPS. The basis of this requirement is that the software design error is a credible source of CMF because software cannot be proven to be error-free. If a postulated CMF could disable any protection function that is required to respond to the design basis event, then a diverse means of effective protection would be necessary. Presented in this paper are the results of the detailed quantitative evaluations which has been performed to confirm the capability of KNGR to cope with non-LOCA events with a postulated CMF in digital PPS.

II. KNGR DESIGN APPROACH FOR DEFENSE-IN-DEPTH AND DIVERSITY

The KNGR digital Instrumentation and Control (I&C) system design related to D-in-D&D is to eliminate predictable CMFs and to obtain high reliability to reduce CMF potential for software in the digital PPS. Predictable CMFs are avoided through seismic and Electro-Magnetic Interference (EMI) qualification, aging analyses and physical separation of equipment. High reliability is realized by deterministic design, simplicity, use of field proven products, a comprehensive verification and validation program, segmentation and diversity. The most severe common mode failure in the digital I&C systems has been found to be a complete malfunction of PPS, which disables reactor trip functions and the actuation of various Engineered Safety Features (ESF). The systems not affected by CMF in digital PPS are 1) Qualified Indication and Alarm System – P (QIAS-P), 2) Main Control Room (MCR) hardwired manual reactor trip, 3) MCR hardwired manual ESF actuation, 4) Information Processing System (IPS) which is a digital based monitoring system, and 5) Alternate Protection System (APS) which is a digital based system to meet Anticipated Transients Without Scram (ATWS) requirements [1]. The control echelon (line of defense) is the non-safety equipment that routinely prevents reactor excursions toward unsafe regimes of operation and is used for normal operation of the reactor. The Nuclear Steam Supply System (NSSS) control systems have D-in-D&D characteristics against CMF in digital PPS because these systems are diverse from digital PPS. In addition, manual operator action is allowed as a diverse means of responding to postulated CMFs if sufficient information and time is available for the operator to detect, analyze and act to mitigate the events with CMF in digital PPS.

III. ANALYSIS METHODOLOGY

Based on extensive qualitative study on all non-LOCA events, five events have been identified as necessary for quantitative analyses to demonstrate the D-in-D&D characteristics of the KNGR design to deal with the CMF in digital PPS. These events include 1) Main Steam Line Break, 2) Total Loss of Reactor Coolant Flow, 3) Control Element Assembly (CEA) Ejection, and 4) Steam Generator Tube Rupture. In addition to these four events, Feedwater Line Break (FLB) was analyzed quantitatively to verify the validity of qualitative evaluation concluding that this event with CMF would not result in the violation of the criteria with respect to primary system pressure boundary. The emphases of the evaluations have been placed on a required action time for plant operators to cope with the events in a manner which preserves core coolability, maintains Reactor Coolant System (RCS) integrity, and prevents excessive offsite doses.

The combinations of design bases events and CMF in digital PPS are categorized as beyond design bases events [2]. Therefore, a best-estimate analysis methodology has been applied according to the regulatory guidance [3]. Major characteristics of the best-estimate analysis methodology include utilizing nominal initial conditions and nominal design data, crediting components and systems being independent and diverse from digital PPS, and crediting appropriate operator actions. In this analysis, the NSSS thermal hydraulic responses are simulated using the CESEC-III computer program [4] while the fuel performances are simulated using the CETOP-D [5] and STRIKIN-II [6] computer codes.

IV. ANALYSES RESULTS AND DISCUSSIONS

<u>Main Steam Line Break Outside Containment</u>

A large energy extraction caused by the steam line break reduces the steam pressure dramatically causing the turbine-generator shutdown terminating the condensate water supply to the feedwater system. The feedwater control system tends to increase the main feedwater flow to the steam generators in response to the decrease in the steam generator water level. All feedwater heating is assumed to be lost immediately due to steam line break. It is also assumed that the feedwater system and feedwater control system are able to maintain the mass of liquid in the steam generators essentially constant until the entire source of main feedwater supply is exhausted. Main concerns for this accident are the maintenance of core coolable geometry, radiological releases, and the primary system integrity.

Figures 1 and 2 show the core power and the RCS pressure variations for the main steam line break event with CMF in digital PPS. A rapid cooldown caused by the steam blowdown through the break causes a sharp increase in the reactor power as shown in Figure 1. Complete depletion of feedwater results in a steam generator dry out. The dry out of steam generators causes a rapid increase in the RCS pressure, which leads to a reactor trip on high pressurizer pressure by the APS at about 900 seconds. The RCS peak pressure is well below 3200 psia, which is adapted as acceptance criteria with respect to primary system integrity. The calculated peak core power and the minimum DNBR are approximately 189% of nominal power and 1.1, respectively. About 1% of fuel failure is predicted to occur as a consequence of the DNB SAFDL violation. The radiological dose release is found to meet the limits specified in 10 CFR 100 guideline. The maximum cladding and fuel centerline temperatures follow the same trend as the power, reaching peak values of less than 670 °F and 4340 °F, respectively. These ensure the maintenance of the core coolable geometry. Diverse systems such as the APS accompanied by intrinsic thermal margin are evaluated to be effective to mitigate the consequences of the main steam line break event with CMF in digital PPS.

Feedwater Line Break

Feedwater line break is initiated by a break in the main feedwater system piping causing the steam generator adjacent to the break to experience a decrease in steam generator inventory and, hence, a reduction in the primary to secondary heat transfer. This leads to a primary system heatup and pressurization. A CMF in digital PPS prevents the reactor trip on high pressurizer pressure. Main concern for this event is whether or not the primary system integrity is maintained.

Figures 3 and 4 show the steam generator inventory and the RCS pressure variations following the feedwater line break with CMF in the PPS. Affected and intact steam generator inventories rapidly decrease and reaches dryout condition within 40 seconds and 170 seconds, respectively. The reduction in the heat removal capability by the secondary system causes the RCS heatup and pressurization. The reactor trip on high pressurizer pressure is triggered by the APS, available during the event. Followed by reactor trip, Pilot Operated Safety and Relief Valves (POSRVs) open to reduce the RCS pressure. The RCS peak pressure is 2650 psia at 22 seconds, which is well below 3200 psia. This ensures the maintenance of the primary system integrity. The APS is effective to mitigate the consequences of the feedwater line break event with CMF in digital PPS.

Total Loss of Reactor Coolant Flow

A complete loss of the forced reactor coolant system (RCS) flow will result from the simultaneous loss of the electrical power to all four Reactor Coolant Pumps (RCPs). The only credible failure which can result in such event is a complete loss of offsite power. As a result of loss of offsite power, all RCPs experience reduction in shaft speed (speed coastdown), causing a rapid deterioration in the reactor core flow (flow coastdown). Due to the CMF in digital PPS, the CPC low RCP shaft speed trip can not be credited in this analysis. The loss of normal electrical power to station equipment results in loss of power to the Control Element Drive Mechanisms. The CEDM motor-generators begin to coast down and an under voltage relay opens the output breaker. This interrupts power to the CEDMs and results in a gravitational drop of the control rods within several seconds following loss of offsite power. Main concern for this event is how much the thermal margin would be degraded due to the coastdown of all four RCP before enough negative reactivity is added by the insertion of control rods by gravity.

Figures 5 and 6 show the core power and the DNBR variations for the total loss of reactor coolant flow event with CMF in digital PPS. The predicted minimum DNBR is 1.75 at 8.4 seconds, which ensures no fuel failure on DNB.

Control Element Assembly (CEA) Ejection

A CEA ejection results from a circumferential rupture of the CEDM housing of the CEDM nozzle. The CEA ejection causes the core power and the RCS pressure to increase rapidly. A CMF in digital PPS prevents the reactor trip on high core power and high pressurizer pressure. Moderator and fuel temperature feedback effects restore the core power to nominal level. The main concern of this event is how much the thermal margin would degrade due to the core power increase and whether the primary system integrity would be maintained. Figures 7 and 8 show the core power and DNBR variations during the CEA ejection event with CMF in digital PPS. Thermal margin behavior depends mainly on the core power variation. The minimum DNBR is well above the DNB SAFDL limit of 1.30, which ensures no fuel failure on DNB. The peak pressure is well below 3200 psia, which is adopted as an acceptance criteria with respect to primary system integrity. The best estimate overpower margin of about 150% for normal operating condition is turned out to play an important role in mitigating the consequences of the CEA ejection event with CMF in digital PPS effectively.

Steam Generator Tube Rupture

The steam generator tube rupture accident is a penetration of the barrier between the RCS and the main steam system. Steam generator tube rupture causes the RCS pressure to decrease due to the leakage of reactor coolant. The probable reactor trip signals are generated by a low hot leg saturation margin or a low DNBR by the Core Protection Calculator (CPC), which is a part of PPS. The CMF in digital PPS prevents these reactor trips from being triggered. The main concern for this event is how much the thermal margin would be degraded due to the decrease in the RCS pressure within a specified time beyond which the operators are assumed to take manual actions to restore the degraded thermal margin. In addition, the radiological release to the atmosphere should be limited within the regulatory guidelines. Figures 9 and 10 show the RCS pressure and DNBR variations for the steam generator tube rupture event with CMF in digital PPS. The RCS pressure decreases continuously. No fuel failure due to DNB is predicted until 30 minutes at which operators are assumed to take manual actions. The radiological release meets the regulatory limit with sufficient margin.

V. CONCLUSIONS

A new systematic best-estimate analysis methodology, which is partly conservative but licensable, for the analyses of non-LOCA events with CMF in digital PPS and applied to the KNGR D-in-D&D analysis. The analysis results demonstrated the capability of the KNGR design to accommodate the non-LOCA events with CMF in digital PPS, which are categorized as beyond design bases accidents. Due to the diverse means (APS and NSSS control systems) to cope with CMF in digital PPS and a sufficient available over-power margin, the consequences are well within the regulatory guidance limits. In addition, the KNGR design offers sufficient safety margin against non-LOCA events without operator actions up to 30 minutes after an event initiation even with CMF. Therefore, the intrinsic D-in-D&D capability of the KNGR design against non-LOCA events with CMF in digital PPS has been verified.

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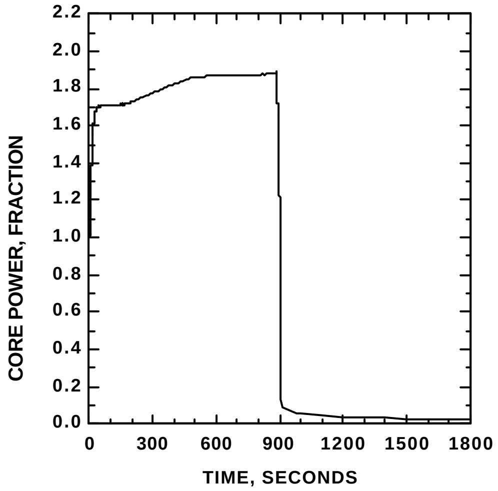

Fig.1 Core Power Transient for Steam Line Break

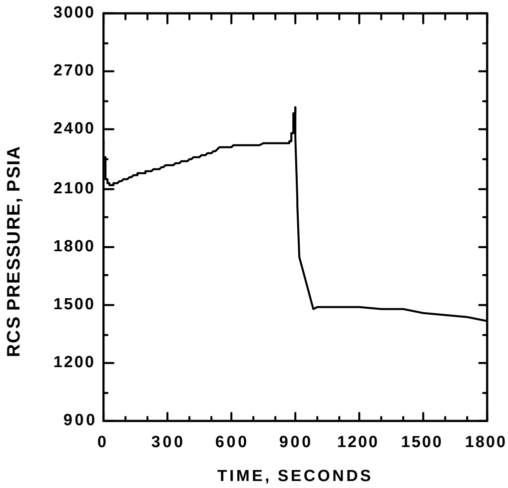

Fig.2 RCS Pressure Transient for Steam Line Break

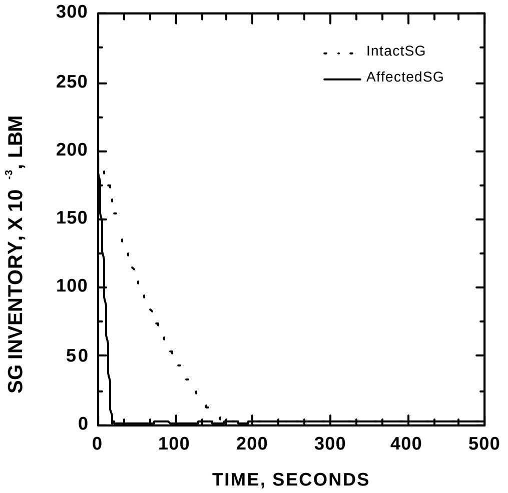

Fig.3 SG Inventory Transient for Feedwater Line Break

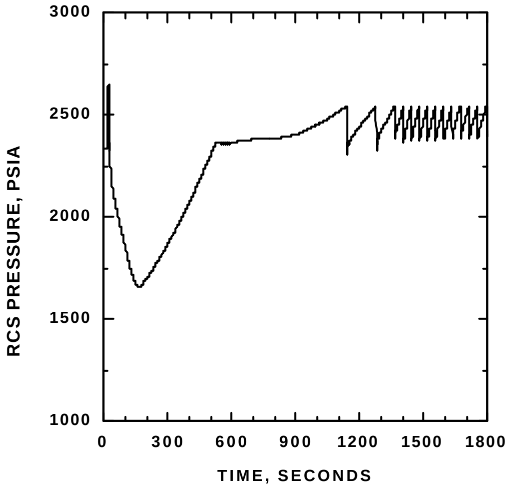

Fig.4 RCS Pressure Transient for Feedwater Line Break

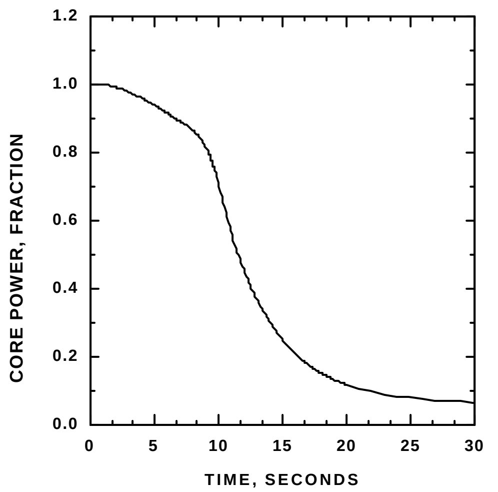

Fig.5 Core Power Transient for Total Loss of Reactor Coolant Flow

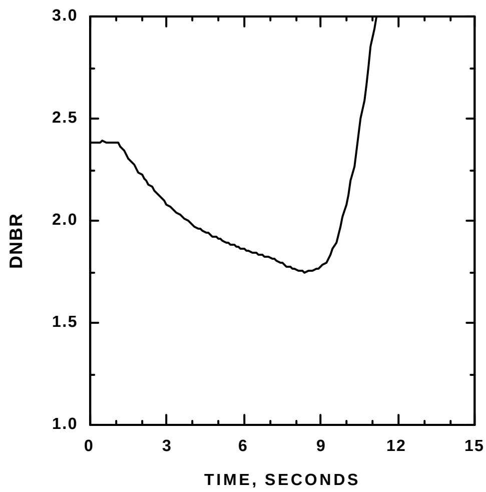

Fig.6 DNBR Transient for Total Loss of Reactor Coolant Flow

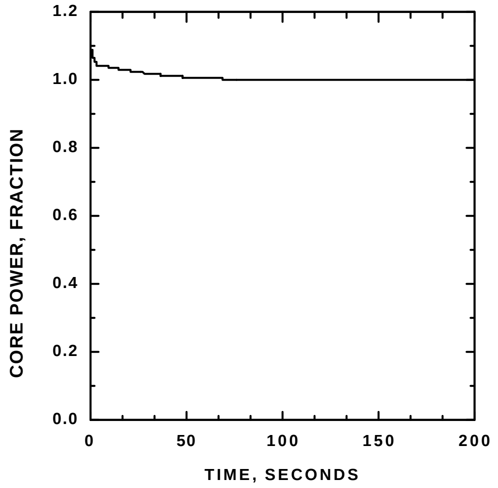

Fig.7 Core Power Transient for CEA Ejection

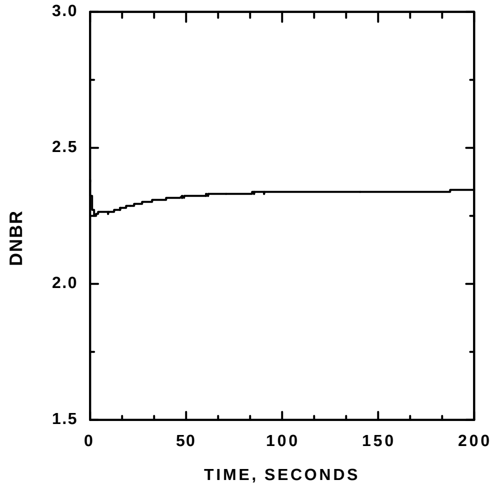

Fig.8 DNBR Transient for CEA Ejection

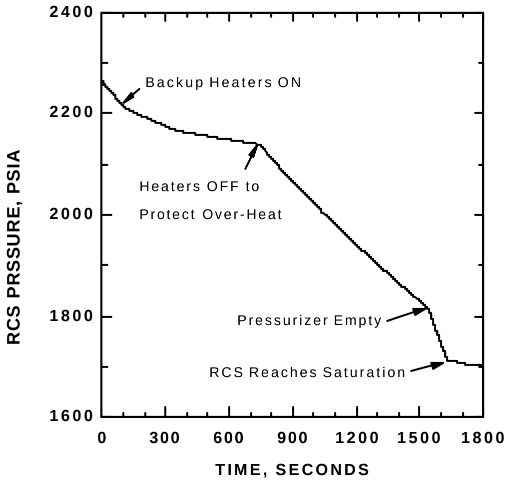

Fig.9 RCS Pressure Transient for Steam Generator Tube Rupture

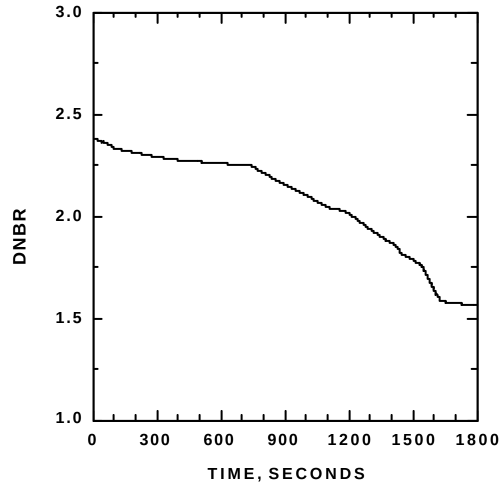

Fig.10 DNBR Transient for Steam Generator Tube Rupture