UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
WASHINGTON, DC 20555-0001

July 24, 2012

NRC INFORMATION NOTICE 2012-12: HVAC DESIGN CONTROL ISSUES CHALLENGE SAFETY SYSTEM FUNCTION

ADDRESSEES

All holders of an operating license or construction permit for a nuclear power reactor or a non-power (research or test) reactor issued under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of and applicants for a power reactor early site permit, combined license, standard design certification, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees about certain events involving heating, ventilation, and air conditioning (HVAC) system design control issues that challenged, or potentially challenged, safety system functions. The NRC expects recipients to review the information contained within for applicability to their facilities and consider actions, as appropriate, to avoid similar occurrences. Suggestions contained within this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Susquehanna Steam Electric Station (Susquehanna) HVAC Controller

On January 3, 2011, PPL, the licensee for Susquehanna, identified a single-point vulnerability in the reactor building HVAC system. The vulnerability was that a failure of a nonsafety-related temperature controller coincident with outside ambient air temperatures below 10 degrees Fahrenheit (^{o}F) could result in a spurious steam leak detection (SLD) system isolation on high differential temperature (ΔT), causing simultaneous isolation of main steam isolation valves (MSIV), the high pressure coolant injection system, and the reactor core isolation cooling system. This vulnerability was common to both Susquehanna Units 1 and 2 and had been in existence since the plants began licensed operations.

PPL initially reported the issue through an event notification (EN) (EN 46519) under 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," as an unanalyzed condition (10 CFR 50.72 (b)(3)(ii)(B)) and an accident mitigation concern (10 CFR 50.72 (b)(3)(v)(D)). However, on February 28, 2011, PPL submitted an updated EN that removed the accident mitigation consideration based on the low likelihood of a reactor

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building temperature controller failure during a period when outside temperature was below 10 °F (both conditions are required for the deficient SLD system isolation on high ΔT to occur). PPL provided additional information pertaining to this issue in the form of a 10 CFR 50.73, “License Event Report [LER] System,” for an unanalyzed condition (LER 3872011001). The LER stated that the single-point vulnerability was discovered during the preparation of a 10 CFR 50.59, “Changes, Tests and Experiments,” determination for an engineering change to remove the SLD high ΔT isolation function to address obsolescence of the function’s components. The licensee attributed the issue to a “less than adequate single-failure analysis performed during the original plant design.”

The original single-failure analysis was performed consistent with accepted practices during the period of the initial plant design. In 2007, Susquehanna engineers received training on failure modes and effects analysis (FMEA) techniques. This training updated the expectations for FMEAs performed on nonsafety systems. Consequently, Susquehanna engineers used the new techniques when evaluating the impact of removing the SLD isolation function and, in the process, identified the single-point vulnerability deficiency.

The corrective actions for this issue included removing the isolation function of the SLD system ΔT instrumentation and performing a FMEA on all nonsafety systems that could cause an isolation of the emergency core cooling system or MSIVs as an extent of condition assessment.

The report, “Susquehanna Steam Electric Station - NRC Integrated Inspection Report 05000387/2011003 and 05000388/2011003 and Exercise of Enforcement Discretion,” dated August 10, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112220409), provides the results of the NRC inspection related to this issue.

Diablo Canyon Power Plant Auxiliary Building Ventilation System Actuation Logic

Diablo Canyon Nuclear Power Plant (DCNPP) completed modifications to its auxiliary building ventilation systems (ABVS) in November 2010. These modifications included replacement of relay-based actuation logic with a programmable logic controller (PLC). The licensee implemented the modification to address problems with reliability and availability (i.e. obsolescence). The licensee reviewed the modification design to ensure applicable single-failure criteria were met. Notwithstanding the licensee’s review, on January 10, 2011, during containment spray pump quarterly testing, a deficiency in the actuation logic of the recently installed PLC resulted in a complete loss of the Unit 2 ABVS when a damper failed to open as required because of leakage past a piston seal. This led one of the two ABVS exhaust fans to trip and prevented the other exhaust fan from starting; thus ABVS became inoperable.

The loss of the ABVS required the licensee to take action in accordance with Technical Specification Limiting Condition for Operation 3.0.3 (i.e., action statement to reduce mode of plant operation) for approximately 20 minutes until operators restored the ABVS system through manual actions. The failure of the piston seal was attributed to using the seal beyond its defined service life, contrary to the requirements of the licensee’s preventive maintenance program for the seal.

DCNPP initially reported this event through a 10 CFR 50.72 EN (EN 46531) as an unanalyzed condition (10 CFR 50.72(b)(3)(ii)(B)) and an accident mitigation concern (10 CFR 50.72(b)(3)(v)(D)). The licensee provided additional information in the form of a 10 CFR 50.73 LER for an unanalyzed condition and safety system functional failure (LER 2752011002). In the LER, the licensee incorrectly attributed the cause of the loss of the

ABVS to a nonconforming single-failure vulnerability in the ABVS system design that existed as part of the original design for both DCNPP Units. It was later determined that the 2010 modifications to the ABVS control logic introduced a single-failure vulnerability, where ABVS exhaust fans tripped when a system damper was not fully opened.

The corrective actions for this issue consisted of modifying the design of both DCNPP units to satisfy the single-failure design criteria, revising the design change process to include a design evaluation of new and old failure modes based on the current licensing and design bases, and revising the licensing basis.

The report, "Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011002 and 05000323/2011002," dated May 11, 2011 (ADAMS Accession No. ML111310608), provides the results of the NRC inspection related to this issue.

Point Beach Nuclear Plant (Point Beach) Control Room Emergency Filtration Fan Thermal Overload

On February 3, 2007, Point Beach lost operability of the control room emergency filtration system (CREFS) because of an inadequately designed modification (LER 2662007001). In October 2006, the licensee installed a modification (high efficiency CREFS fan motors) for the purpose of increasing the low flow margin. During the design of this modification, an incorrect assumption was made that outside temperature had a negligible effect on motor current draw, so no compensation for low temperature was included in the motor thermal overload design. On February 3, 2007, with outside temperature at 6 ^{o}F, a CREFS fan tripped during a Technical Specification surveillance test because of a thermal overload relay trip. After evaluating the cause of the trip, the licensee declared both CREFS fans inoperable because the fan motors had inadequately sized thermal overload heater elements.

The corrective actions for this issue included replacing the overload heater elements with elements having trip current setpoints adjusted to values that considered design requirements.

The report, "Point Beach Nuclear Power Plant, Units 1 and 2, NRC Integrated Inspection Report 05000266/2007002 and 05000301/2007002," dated April 12, 2007 (ADAMS Accession No. ML071020081), provides the results of the NRC inspection related to this event.

BACKGROUND

Criterion III of Appendix B to 10 CFR Part 50 requires, in part, that licensees ensure that applicable regulatory requirements and design basis are "correctly translated into specifications, drawings, procedures, and instructions." Furthermore, "design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design..."

DISCUSSION

In each event described in this IN, a safety system's function was challenged or potentially challenged because of design control issues. In the first case, a long-standing design control issue was finally identified after the licensee adopted updated methods of analyzing nonsafety system designs for single failures. In the second and third cases, actual safety system functional failures occurred as a result of licensees implementing deficient modifications. These events illustrate the importance of evaluating modifications rigorously to verify that design-basis requirements are satisfied.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or to the appropriate Office of Nuclear Reactor Regulation or Office of New Reactors project manager.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library/Document Collections.

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