UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

July 22, 2004

NRC INFORMATION NOTICE 2004-15: DUAL-UNIT SCRAM AT PEACH BOTTOM UNITS 2 AND 3

Addressees:

All holders of operating licenses for nuclear power reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed from the reactor vessel.

Purpose:

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to recent experience in which a dual unit facility lost offsite power, had a dual unit scram, and experienced other problems including the loss of a common emergency diesel generator (EDG). It is expected that recipients will review this information for applicability to their facilities and consider actions, as appropriate. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On September 15, 2003, offsite power to the emergency buses at Peach Bottom Units 2 and 3 was lost for about 16 seconds when two of the three offsite power sources were briefly lost. All four EDGs automatically started and supplied power to the emergency buses. The third offsite power source remained available to two of the four plant non-emergency plant buses throughout the event.

The offsite power grid dispatcher notified the control room that the portion of the offsite power that was supplying the emergency buses was available half an hour after the event started. However, because the emergency buses were powered from the EDGs and plant transient response actions were the operational priority, operators did not transfer from the EDGs to offsite power for several hours until they were more certain of the reliability of the offsite power source. The licensee determined that the loss of offsite power was the result of a lightning strike approximately 35 miles northeast of the site.

Before the event, Unit 2 was operating at full power and Unit 3 was operating at 91 percent of full power. Both units automatically scrammed when power was lost to the reactor protection system motor generator sets. Containment isolation signals resulted in the closure of the main steam isolation valves and isolation of each reactor from its normal heat sink, the condenser.

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All four EDGs automatically started and supplied power to the emergency buses; each EDG supplies power for two buses (one per unit). The licensee was able to safely bring both units into the cold shutdown condition. However, the shutdown of each unit was complicated both by equipment challenges and by procedural problems. The NRC organized an Augmented Inspection Team (AIT) because of the overall risk significance of the event and multiple failures in systems used to mitigate the event. The AIT mission was to determine the causes, conditions, and circumstances relevant to issues directly related to the event and to assess the safety significance of the event (NRC Augmented Inspection Team Report 05000277/2003013 and 05000278/2003013, ADAMS Accession No. ML033530016).

Discussion:

The most significant equipment problem during this event was the unexpected E2 EDG trip during the cooling of the Unit 2 torus while other EDGs were supplying power to the emergency buses. The E2 EDG shut down due to an engine protective trip initiated by low jacket water pressure. The AIT found that combustion gases entered the jacket water coolant system because of one or more leaking cylinder adapter gaskets, causing low jacket water pressure and automatic shutdown of the E2 EDG. The leakage was due to deficient installation procedures and stress relaxation of the cylinder adapter gaskets. These adapter gaskets, made of copper, provide a seal between high-pressure gases in each cylinder and the jacket water system. The licensee concluded that the root cause was inadequate initial pre-loading combined with the natural stress relaxation of the copper over time. The licensee has four Fairbanks Morse 12 cylinder, opposed piston diesel engines for both units.

The AIT found that the EDG cylinder liner replacement procedure did not incorporate adequate guidance to ensure proper sealing of the cylinder liner adapter gaskets. The gaskets relaxed over several years, allowing combustion gases to enter the jacket coolant system. Additionally, the licensee may have missed opportunities associated with jacket water anomalies. Degraded conditions, such as jacket water leaks and high vibration on the E2 EDG from 1996-2002, were tolerated and a condition adverse to quality following two instances of low jacket water pressure was not corrected.

The licensee performed a number of corrective actions to remedy the EDG gasket problem. The licensee replaced all adapter gaskets on the tripped EDG, inspected the cylinders during hydrostatic testing, temporarily installed a sight glass to ensure no combustion gas leakage, revised test and maintenance procedures, and sampled the expansion tank air space and jacket coolant heat exchanger for combustion gases. The final three actions were performed on all the EDGs.

The AIT found that the maintenance procedure for installing the cylinder liner adapter gaskets on the EDGs was deficient and that the licensee took inadequate corrective actions for the low jacket water pressure conditions observed on the E2 EDG in March and April 2003. Using the reactor safety Significance Determination Process (SDP), the AIT determined this incident to be a White finding for Unit 2 (i.e., a low-to-moderate safety-significant finding that may require additional NRC inspection) and a Green finding for Unit 3. The difference in risk significance between the units is due to differences in electrical bus loads.

The Unit 2 transient was complicated by the following factors:

1. Due to the momentary loss of offsite power, the controlling channel of the Unit 2 condenser hotwell level instrumentation failed low. This previously identified equipment deficiency resulted in the draining of the Unit 2 condensate storage tank to the Unit 2 condenser hotwell. The condensate storage tank is the preferred common suction of two Unit 2 mitigating systems, the high-pressure coolant injection system and the reactor core isolation cooling system. As a result of the decreasing condensate storage tank level, the suction of these mitigating systems automatically but unexpectedly changed from the Unit 2 condensate storage tank to the Unit 2 torus.

2. As a result of the transient, the Unit 2 torus water heated up, necessitating the use of residual heat removal (RHR) pumps to cool the Unit 2 torus. At the Peach Bottom site, there are 4 RHR pumps per unit and 4 EDGs common to both units. Therefore, 1 RHR pump from each unit is associated with 1 EDG. A minimum of 1 of the 4 RHR pumps per unit is required to satisfy the containment cooling design function. The licensee needed to use a minimum of 1 RHR pump on both units but was prohibited from simultaneously using pumps powered by the same electrical division, that is, off the same EDG. Thus the Unit 2 A RHR pump and the Unit 3 A RHR pump could not be used at the same time. This is due to electrical load restrictions on the EDG that supplies the same electrical division for both Units 2 and 3. This is a known design limitation of the Peach Bottom station involving the significant electrical load requirements for operating the RHR pump motors.

3. On isolation of the Unit 2 condenser hotwell due to closure of the main steam isolation valves, the associated E2 EDG unexpectedly tripped, stopping the Unit 2 torus cooling. The E2 EDG tripped on low jacket water coolant pressure, which stopped the inservice RHR pump and drained the B torus cooling loop, reducing the availability of torus cooling on Unit 2.

4. A number of other deficiencies complicated operator response and recovery actions.

The Unit 3 transient was complicated by several different factors:

1. The Unit 3 D safety relief valve opened as designed on high reactor pressure but failed to close at the appropriate decreasing reactor pressure setpoint. Over the next 15 minutes, reactor pressure decreased to 369 psig before the valve closed, which allowed injection by condensate pumps and an increase in reactor water level to the high-level setpoint before operators manually tripped these pumps. The valve closed with no operator action. The cause of the initial failure of the valve to close was determined to be pilot valve leakage.

2. The Unit 3 G safety relief valve initially opened automatically on high reactor pressure as designed and was subsequently remotely operated to control reactor pressure. However, on a reactor pressure control operation much later in the event, the valve failed to open on demand from the main control board control switch. The cause of the failure

of the valve to open was determined to be steam leaking through the valve packing into the air operator. The steam damaged the diaphragm of the air operator and prevented the valve from manually operating.

3. The Unit 3 D outboard main steam isolation valve failed to close upon receipt of the Group I isolation signal, remained open for 76 minutes, and then closed with no operator action. The redundant inboard main steam isolation valve appropriately closed as designed.

4. A number of other deficiencies complicated operator response and recovery actions.

This information notice requires no specific action or written response. If you have any questions about information in this notice, please contact one of the technical contacts listed below or the appropriate NRR project manager.

/RA/
William D. Beckner, Chief
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Division of Inspection Program Management
Office of Nuclear Reactor Regulation

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2004-14	Use of less than Optimal Bounding Assumptions in Criticality Safety Analysis at Fuel Cycle Facilities	07/19/2004	All licensees authorized to possess a critical mass of special nuclear material.
2004-13	Registration, Use, and Quality Assurance Requirements for NRC-Certified Transportation Packages	06/30/2004	All materials and decommissioning reactor licensees.
2004-12	Spent Fuel Rod Accountability	06/25/2004	All holders of operating licenses for nuclear power reactors, research and test reactors, decommissioned sites storing spent fuel in a pool, and wet spent fuel storage sites.
2004-11	Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzle	05/06/2004	All holders of operating licenses or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2004-10	Loose Parts in Steam Generators	05/04/2004	All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

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