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O-P1-17

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

August 18, 1998

NRC INFORMATION NOTICE 98-31: FIRE PROTECTION SYSTEM DESIGN DEFICIENCIES AND COMMON-MODE FLOODING OF EMERGENCY CORE COOLING SYSTEM ROOMS AT WASHINGTON NUCLEAR PROJECT UNIT 2

Addressees

All holders of operating licenses for nuclear power reactors, except those licensees that have permanently ceased operations 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 a rupture of a fire water system valve, due to a water hammer, in a fire main vertical riser at Washington Nuclear Project Unit 2 (WNP-2) that flooded two emergency core cooling system (ECCS) equipment rooms. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On June 17, 1998, WNP-2 was in Mode 4 (cold shutdown) and preparations were underway for a plant startup. At approximately 1:45 p.m., multiple fire alarms were received in the control room coincident with three main fire pumps automatically starting and several loud water hammer noises being heard throughout the plant. The water hammer caused a fire protection isolation valve (FP-V-29D) to rupture in the fire protection system riser in the northeast stairwell of the reactor building. Water from the stairwell entered residual heat removal (RHR) pump room C through a watertight door that had not been adequately secured and began rapidly flooding the room. A reactor drains system valve (FDR-V-609) located in a line connecting the sumps of the RHR C and low-pressure core spray (LPCS) pump rooms failed to close as designed and allowed water to flow into the LPCS pump room. The flood water completely submerged the RHR C pump and motor and the Division II keepfill pump, which serves RHR B and C and is also located in the room. Water in the LPCS pump room rose to a level just below the pump motor and also completely submerged the minimum flow valve and the Division I keepfill pump, which serves both the LPCS and RHR A trains.

To isolate the flooding, plant operators secured the fire pumps, which impaired the normal fire suppression capability of the station. On the basis of these events, the plant operators declared a notification of unusual event (NOUE) and activated the plant emergency response organization. As a compensatory measure for the loss of the normal fire suppression capability, the nearby Hanford fire department dispatched emergency equipment to the site. Subsequently, the plant staff placed the fire suppression system in an alternate configuration, which was less susceptible to water hammer and terminated the unusual event.

Discussion

I. Fire Protection System Design and Operation

The fire protection system at WNP-2 consists of two diesel-driven and two electric-driven fire pumps. The two electric pumps and one of the diesel pumps have a capacity of 2000 GPM each and draw a supply from the circulating water basin. The remaining diesel-driven pump has a capacity of 2500 GPM and is supplied by a 400,000-gallon embankment-supported Fabritank (i.e., bladder). The fire pumps are normally in standby and the system pressure is maintained at approximately 150 psig by a 220-GPM jockey pump. The system is arranged such that the pumps supply a main header, which, in turn, supplies various yard hydrant isolation valves and building standpipes. The risers in the reactor building are the high points of the fire main system at WNP-2 and rise approximately 180 feet above the main yard loop.

Additionally, the design also includes a number of PRE-ACTION systems, which are not normally filled with water. Upon actuation of the associated detector(s) for a given PRE-ACTION system, the PRE-ACTION system valves will open and allow water to flow from the main header into the associated piping. Some of the plant PRE-ACTION systems are activated by ionization-type detectors, whereas other PRE-ACTION systems rely on thermal detectors for actuation. However, the sprinkler heads associated with the downstream piping are not actuated during a PRE-ACTION unless the thermal-fusible links are melted on the individual heads, thereby completing the flow path for the fire water.

This event was initiated by the actuation of fire detectors during cutting and grinding activities, which were taking place in the diesel generator building. The fire detectors, sensing the smoke from the maintenance activities, activated a fire protection PRE-ACTION station, which caused the associated PRE-ACTION valves to open and fill the normally dry sprinkler line header. (A second PRE-ACTION station also actuated due to sympathetic effects.) However, no actuation of the associated sprinklers occurred since they are ultimately initiated by thermal-fusible links. The depressurization of the fire water system during the filling of the PRE-ACTION lines caused significant voiding in the upper portions of the reactor building vertical fire main risers and generated an auto-start signal for all four main fire water pumps to start on low system pressure. Three of the pumps started immediately, and the fourth pump began a 30-second time delay sequence for starting. The concurrent operation of the three pumps resulted in a rapid reflood of the reactor building risers and collapsed the void that had been created in the northeast stairwell riser. This sequence of events caused a significant water hammer that ruptured a 12-inch, cast-iron, fire protection system isolation valve that was located in the stairwell riser. The licensee determined that the design of the fire protection system was inadequate in that the system is configured such that destructive water hammer forces are generated during anticipated transients when the system is in a normal lineup. Specifically, the significant voiding caused by the PRE-ACTION actuation, coupled with the simultaneous starting of the main fire pumps and the unfavorable geometry of the reactor building riser and its associated supports, contributed to the severity of this event.

II. Common-Mode Flooding Considerations

During the event, approximately 163,000 gallons of water were introduced into the northeast stairwell, RHR C, and the LPCS pump rooms. Additionally, some minor leakage of flood water occurred between the LPCS pump room and the vestibule separating that room from the adjacent high pressure core spray pump room. Water also leaked from the RHR C pump room into the adjacent reactor core isolation cooling room through a double watertight door arrangement that separates those two rooms.

With respect to the flooding of the ECCS rooms, it was determined that the door to the RHR C pump room was left in an unsecured condition sometime before the event. A review of a door alarm printout from the common alarm station indicated that the door had changed state several minutes before the fire protection system rupture. With the door in an unsecured or open condition, an unrestricted pathway existed for flood water to flow from the stairwell into the RHR C pump room. However, it was noted that the watertight doors for the northeast stairwell access to the RHR C pump room and the LPCS pump room were not designed or installed to prevent flooding from the stairwell from entering the associated pump rooms (the doors were designed to seal from inside the rooms). Thus, even if the door had been secured, water would have entered the pump rooms, albeit at a much slower rate. The licensee's flooding analyses had assumed that a stairwell flood would eventually render the LPCS and RHR C systems inoperable and that operator actions to start the RHR A and B trains would be taken before the loss of the keepfill pumps for those systems that are located in the LPCS and RHR C pump rooms.

The floor drains for the RHR C pump room and the LPCS pump room drain to the same sump. A single isolation valve, FDR-V-609, located in the drain line, is designed to isolate the LPCS pump room drain from the RHR C pump room drain in the event that the sump is overfilled. This nonsafety-related isolation valve is air operated via a four way shuttle valve and accumulator. During normal operation the isolation valve is opened and closed by supplying or removing air pressure to the four way shuttle valve which ports the air to the isolation valve operator and the accumulator. The system is designed such that the isolation valve should fail closed on a loss of air supply pressure. This air supply is controlled (on/off) by a solenoid operated valve, FDR-SPV-609 which is upstream of the four way shuttle valve. The licensee believes that during the flooding event the solenoid operated valve failed to fully close which resulted in a reduction of pressure in the accumulator (resulting in a lower force to drive the sump isolation valve closed) and a failure to vent the air supply line to atmosphere (higher pressure in the supply line opposes closing of the sump isolation valve). As a result, the sump isolation valve failed to close automatically when the sump reached its high level trip point. This allowed water from the RHR C pump room to flow through the 3-inch sump cross-connect piping and into the LPCS pump room flooding it to a level just below the pump motor. Plant operators were unable to close the isolation valve manually from the control room. The licensee's preliminary failure analysis indicated that the solenoid operated valve, an ASCO model #WJNP831654E, likely failed to operate due to age hardening of the Buna-N diaphragm. The licensee believes that this diaphragm has not been replaced since initial plant construction. The licensee was continuing its failure analysis at the time this Information Notice was issued.

III Licensee Corrective Actions

As an immediate corrective action, the licensee pumped the water from the flooded areas. The fire protection system was returned to a functional (but degraded) status by isolating the ruptured valve and returning the PRE-ACTION system to its normal condition. Subsequently, the licensee repaired or replaced all affected components. The ruptured 12-inch cast-iron valve was replaced with a cast-steel valve. As interim corrective actions, the licensee has established a nitrogen bubble at the top of both the fire water system risers in the reactor building to provide a cushioning effect, and is maintaining two fire pumps in continuous operation in order to avoid the significant voiding expected during postulated PRE-ACTION scenarios. The licensee briefed NRC management on the corrective actions at a public meeting in the Region IV offices in Arlington, Texas, on July 2, 1998. The licensee committed to long-term corrective actions, which included reviews to determine if the flooding analysis in the final safety analysis report is adequate and whether the floor drain valves and door seals meet design requirements. Additionally, the licensee is reviewing potential design changes for the fire protection system to eliminate the susceptibility to water hammer. The licensee restarted the unit on July 3, 1998.

An NRC augmented inspection team (AIT) was on site from June 17 to 23, 1998. The results of the AIT were presented at a public exit meeting on site on July 8, 1998, and were documented in NRC Inspection Report 50-397/98-16, which was issued on July 17, 1998. Preliminary Notification of Occurrence PNO-IV-98-026, which described this event, was issued on June 18, 1998, updated on June 19, 1998, and updated again on June 23, 1998.

This information notice requires no specific action or written response. However, licensees are expected to review the information provided to determine whether similar system vulnerabilities exist at their facilities. Additionally, recipients are reminded that they are required by 10 CFR 50.65 to take industry-wide operating experience (including the information presented in NRC information notices) into consideration, where practical, when setting goals and performing periodic evaluations. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) Project Manager.

for

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Office of Nuclear Reactor Regulation

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98-28	Development of Systematic Sample Plan for Operator Licensing Examinations	8/3/98	All holders of operating licenses for nuclear power plants
98-27	Steam Generator Tube End Cracking	7/24/98	All holders of operating licenses for pressurized-water reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed for the reactor vessel
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OL = Operating License
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orig /s/'d by C.I. Grimes
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