April 2, 1999

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: REQUEST FOR A TECHNICAL REVIEW OF A DRAFT INFORMATION NOTICE DESCRIBING THE UNANTICIPATED REACTOR WATER DRAINDOWN AT QUAD CITIES NUCLEAR POWER STATION, UNIT 2, ARKANSAS NUCLEAR ONE, UNIT 2, AND THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT

Dear Mr. Hutchinson:

The U.S. Nuclear Regulatory Commission is planning to issue an Information Notice (IN) that describes the unanticipated reactor water draindown at Quad Cities Nuclear Power Station, Unit 2, Arkansas Nuclear One, Unit 2, and the James A. FitzPatrick Nuclear Power Plant. This IN is being issued to alert other licensees as to the potential for such an event happening at their facilities.

We request that you review the enclosed draft IN to ensure that the technical information regarding the event at your plant is accurate. If we do not receive written comments within 1 week of the date of issuance for this letter, we will assume that you have no comments. Your cooperation in this matter is appreciated.

Sincerely,

ORIGINAL SIGNED BY

M. Christopher Nolan, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-265, 50-368, and 50-333

Enclosure: As stated

cc: See next page

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Mr. C. Randy Hutchinson
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 2

cc:

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

March XX, 1999

NRC INFORMATION NOTICE 99-XX: UNANTICIPATED REACTOR WATER DRAINDOWN AT QUAD CITIES NUCLEAR POWER STATION, UNIT 2, ARKANSAS NUCLEAR ONE, UNIT 2, AND THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT

Addressees

All holders of licenses for nuclear power, test, and research reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to the potential for personnel errors during infrequently performed evolutions that result in, or contribute to, events such as the inadvertent draining of water from the reactor vessel during shutdown operations. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to prevent a similar occurrence. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response to this notice is required.

Description of Circumstances

Quad Cities, Unit 2

On February 24, 1999, Quad Cities Unit 2 was in cold shutdown with reactor water temperature at about 144 °F and reactor water level in a band of 90 to 94 inches indicated level (normal level during operations is 30 inches indicated or about 173 inches above the top of active fuel [TAF]). Core cooling was being maintained in a band of 120 °F to 170 °F by the "A" loop of the residual heat removal (RHR) mode of shutdown cooling, after being switched from the "B" loop at about 12:32 a.m. Sometime later operators noted a decreasing reactor water level and at about 1:02 a.m. secured the "2A" RHR pump and isolated shutdown cooling. At 1:55 a.m. operators restored the "2A" loop of shutdown cooling to the proper lineup and started the "2A" RHR pump. Water level had decreased to a minimum of about 45 inches indicated, and reactor water temperature had risen to a maximum of about 163 °F. Forced circulation of reactor vessel water using a reactor recirculation pump remained in effect throughout the event.

On the basis of post event reviews, it appears that the minimum flow valve was left open because the nuclear station operator failed to ensure that the tasks were performed in the sequence specified in the operating procedures. The nuclear station operator who was directing the evolution from the control room gave the non-licensed operator permission to de-energize the breaker for the "A" RHR minimum flow valve operator before the valve was taken to the required closed position. De-energizing the breaker also removed power to the valve

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position indicator lights in the control room. Thus, when the nuclear station operator tried to verify that the valve was closed, there was no position indication in the control room to make that verification. The nuclear station operator made the incorrect assumption that the valve was already closed and moved to the next step in the procedure. This failure to close the "A" RHR minimum flow valve opened a drain path from the reactor to the suppression pool. To further complicate the event, the operating crew did not recognize that there was any problem until the water level had decreased about 13 inches. After detecting the decrease, the operating crew was slow to react, which allowed the level to decrease another 20 inches before the operators isolated shutdown cooling which terminated the draindown. The licensee estimated that a total of 6000 to 7000 gallons was drained from the reactor to the suppression pool.

Poor practices in operations including poor communications, poor activity briefings for high-risk activities, lack of pre-shift briefings, inadequate supervision of important control room activities, inadequate monitoring of control room panels, and slow event response may have contributed to the event. However, although the unintended loss of inventory to the suppression pool was significant and highlighted significant weaknesses in plant operations, the safety significance was minimized by two features. First, a reactor recirculation pump remained in service throughout the event which served to distribute decay heat. Additionally, an automatic isolation of shutdown cooling would have occurred at 8 inches indicated level which would have stopped the draining event. An indicated water level of 8 inches corresponds to approximately 151 inches of water level above the TAF in the reactor core.

Arkansas Nuclear One, Unit 2

On February 2, 1999, at Arkansas Nuclear One, Unit 2, the operators were draining the refueling canal in preparation for installing the reactor vessel head. Refueling was complete and steam generator nozzle dams were installed. The operators were using the two low pressure safety injection (LPSI) pumps to drain the canal to the refueling water storage tank; one pump also served as the shutdown cooling pump. The rate of draindown was approximately 3.3 inches per minute. When the water level reached 105 inches, the reactor operator noted that level started to lower rapidly. Operators stopped one of the LPSI pumps and instructed a local operator to close the isolation valve to the refueling water tank. This manually operated valve required 55 turns of the handwheel to fully close. Within approximately 1.5 minutes, the reactor vessel level had dropped below the 65 inch level (where reduced inventory begins) and continued down to 56 inches before the valve could be fully closed. (Reference zero on these level instruments is the bottom of the hot leg, with mid-loop being defined at approximately 24 inches.) The average rate of level decrease between 105 inches and 56 inches was 33 inches per minute. At its lowest level, 56 inches indicated, there were still 93 inches of water above the TAF. Using the high pressure safety injection (HPSI) pump the operators brought the level back up to 90 inches. The plant was in reduced inventory operations (below 65 inches) for approximately 7 minutes. During the event the level remained well above the point where LPSI pump cavitation would be expected.

On the basis of post event reviews, it was determined that the procedure used for draining down the refueling canal was inadequate in that it provided the wrong level at which operators were to secure the draining. The procedure incorrectly stated that the draindown should be secured at the 90-inch level. The procedure should have directed that the rate of draining be secured at the 106-inch level so that appropriate precautions could be taken before resuming the draindown. These precautions should have included reminders to the operating crew that below the 106-inch level the level will drop much more quickly because most of the water has been drained from the refueling canal. Therefore, in order the maintain control of the water level, the draindown rate should be decreased and an operator should be stationed to directly monitor the level.

Additional factors that contributed to this event include: the operators received little specific training on this evolution, and the crew was inexperienced in performing this task; the task should have been classified as an infrequent task, requiring a more thorough briefing; and, operators failed to follow the procedure that required that an operator be stationed to monitor the refueling canal level and relied instead on a camera that did not provide a clear picture of the water level in the refueling canal.

FitzPatrick

On December 2, 1998, at the James A. FitzPatrick Nuclear Power Plant, the operators were in the process of reassembling the reactor following refueling. Operators were controlling the reactor vessel water level at approximately 350 inches above TAF by adjusting the water discharge rate to compensate for the constant input from the control rod drive cooling water system. As required by their risk analysis, they were relying on two independent reactor level instruments. One was the wide-range level indicator (which provided indication up to the top of the reactor vessel) and the other was a narrow-range indicator which was off-scale high.

In order for the wide-range level indicator to remain operable with the reactor head removed, a temporary standpipe and fill funnel were used to replace a portion of the reference leg. At the time of the event, the licensee was in the process of removing this temporary standpipe and reinstalling the original reference leg components. As the water drained from the standpipe, it caused the wide-range level indicator to erroneously show an increasing water level. For a period of approximately one hour, the operators in the control room, unaware that the ongoing maintenance would cause an error in the indicated water level, compensated for the apparent increasing level by increasing the discharge rate. This action had the effect of reducing the actual water level from 350 inches to 250 inches. During the same time period, the operators were also in the process of filling and venting the reactor feedwater piping, which also affected the reactor water level. Once the normal reference leg piping had been reinstalled and the reference leg began to refill, the indicated level decreased from 350 inches to the actual level of 250 inches. The second level instrument (the narrow-range level indicator) which does not come on-scale until the level goes below 224 inches, remained off-scale high.

When operators discovered the level discrepancy, they used a temporary pressure gauge connected to the reactor vessel low-point tap to confirm the actual water level. After confirming the accuracy of the wide-range indicator, they restored the level to 350 inches. The 100-inch error represented approximately 15,000 gallons of water. The licensee determined that the safety significance of this event was low since the reactor was in cold shutdown and the reactor water level remained well above the TAF. In addition, the drain-down would have been limited by an automatic isolation of the draindown path, which would have occurred at 177 inches above the TAF.

The licensee's review of the event identified weaknesses in the operator's knowledge of the reactor assembly process and weaknesses in the plant risk assessment process. Contrary to the assumption in the plant risk analysis that two reactor water level indicators were available, only one, the wide-range indicator, was able to provide level indication above 224 inches. When the reference leg on the wide-range instrument was disassembled and drained, it was rendered inoperable. The second instrument, the narrow-range indicator, was pegged off-scale high and remained that way throughout the event because the level never dropped below 224 inches. Proposed corrective actions included procedural enhancements to address the loss of level indication during reactor disassembly and reassembly and providing for an alternate means of level indication.

Discussion

Personnel errors appear to have caused, or contributed to, these three inadvertent reactor vessel draindown events. The likelihood of personnel errors is dependent upon the operator's knowledge of the task gained through previous experience and training. It is also dependent upon the quality of the procedures used to perform the task, the level of supervision, the adequacy of pre-job briefings, fatigue, and distractions resulting from multiple tasks. In each of the events, the plant staff made errors during a seldom-performed evolution. Because it was a seldom-performed evolution, more training, better pre-job briefings, closer supervision, and procedures that contain more details than those for frequently performed activities might have prevented the event.

The current trend in the Nuclear Power industry is to reduce cost by reducing plant staff and minimizing outage time. These reductions in plant staff may have led to an overall lowering of plant staff knowledge due to the loss of some of the most experienced personnel, including supervisors and managers, at many plants. It is possible that decreases in plant staff experience levels would show up first in infrequently performed tasks such as outage activities, rather than in routine tasks performed during power operation. In addition, the decrease in plant staff and shorter outage times could have the combined effect of increasing the workload of the remaining staff. Increases in work load also leads to longer shifts, fatigue and an increase in the likelihood of personnel errors.

The extent to which these factors may have contributed to the events described in this information notice is unknown. However, these events should serve as a reminder to all licensees that reductions in staff and decreases in outage times can lead to an increased number of personnel errors if they are not carefully controlled.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below, the appropriate regional office, or the appropriate office of Nuclear Reactor Regulation (NRR) Project Manager.

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical contact: Chuck Petrone, NRR
301-415-1027
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REFERENCES:

NRC Integrated Inspection Report No. 50-333/98-08, issued February 10, 1999 (Accession No. 9902170348) for the James A. FitzPatrick Nuclear Power Plant for the period November 22, 1998, through January 10, 1999.

Attachments:
1. List of Recently Issued NMSS Information Notices
2. List of Recently Issued NRC Information Notices