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0CAN120501

December 19, 2005

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Denial of Non-Cited Violations and Finding in NRC Integrated Inspection Report 50-313/2005-04 and 50-368/2005-04
Arkansas Nuclear One – Units 1 and 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6

Dear Sir or Madam:

NRC Inspection Report 2005-04 for Arkansas Nuclear One (ANO) issued on November 7, 2005, summarized the results of the Integrated Inspection for the third quarter of 2005. Per 10CFR50.4, and in accordance with the guidance in the Enforcement Policy, Entergy hereby disputes three of the findings identified in the report along with the cross-cutting aspects of these findings. A detailed assessment of these findings is presented in the attachment.

A "Green" Finding (FIN) was identified concerning the failure to adequately scope the effects of maintenance on pressurizer level instrumentation. This finding was determined to be more than minor because it affected the human performance attribute under the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions; however, our evaluation of the finding concluded that the finding should have been characterized as minor.

Entergy agrees this condition occurred as a result of an improperly scoped trouble shooting plan and corrective actions are being taken to address this deficiency. However, it is Entergy's contention that this event did not result in an upset in plant stability and did not challenge safety systems since plant parameters never left the normal operating band and the unplanned increase in Reactor Coolant System (RCS) pressure and reactor power would have been terminated by normal control systems even if operator action had not been taken. Therefore, it did not impact the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions.

IE06

A "Green" Non-Cited Violation (NCV) of 10CFR50.65(a)(4) was identified for the failure to adequately assess risk for an isolated electromatic relief valve (ERV). Entergy denies that a violation of 10CFR50.65(a)(4) occurred. Our attached evaluation has concluded that an adequate risk assessment was performed when isolating the ERV and during subsequent maintenance evolutions.

A "Green" NCV containing a cross-cutting element of human performance was identified concerning inadequate procedures which lead to reactor coolant pump (RCP) seal damage. The inspectors considered that the failure to have an adequate procedure for ensuring isolation of seal injection when a Unit 2 RCP was uncoupled was a performance deficiency. The inspectors determined that this finding is greater than minor because it was associated with the mitigating systems cornerstone configuration control attribute and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

Entergy disagrees that this condition constitutes a performance deficiency; therefore, a violation of Unit 2 Technical Specification 6.4.1 did not occur. Entergy had no foreknowledge of the need for procedural controls to prohibit applying seal injection to an uncoupled RCP. Furthermore, if the condition were considered a performance deficiency, it should not be considered more than minor given the mitigating system cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was not affected.

Corrective action is being taken to ensure seal damage does not occur in the future due to applying seal injection to an uncoupled RCP. Additionally, Entergy has issued an operating experience report (OE-20704) to make other licensees aware of this vulnerability.

The Reactor Oversight Program is an effective process to monitor and document nuclear power plant performance and provides a useful representation for public review. It is critical that findings be appropriately characterized so that the public is provided an accurate view of licensee performance. Characterizing minor events such as the subject findings of this submittal as "greater than minor" could "skew" performance data and lead to a public misconception of actual nuclear power plant safety performance.

Entergy has entered each of these conditions into our corrective action program where action will be taken to address deficiencies. However, as discussed in the attachment, Entergy does not agree with the NRC categorization of these conditions. Therefore, Entergy respectfully requests that the NRC withdraw the three findings along with the associated cross-cutting aspects.

Additionally, the guidance in the inspection report afforded 30 days of the date of the inspection report to provide a response if any issues identified in the report were contested. On December 6, 2005, in conversations with NRC Region IV Branch Chief, David N. Graves, a verbal due date extension to December 19, 2005, was granted.

There are no new commitments contained in this submittal.

Sincerely,

for J. Forbes

JSF/SLP

Attachment: Denial of Non-Cited Violations and Finding in NRC Integrated Inspection Report 50-313/2005-04 and 50-368/2005-04

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Attachment 1

0CAN120501

Denial of Non-Cited Violations and Finding in
NRC Integrated Inspection Report 50-313/2005-04 and 50-368/2005-04

Finding 50-368/2005004-05

Excerpt from NRC Inspection Report 2005-04 for ANO issued on November 7, 2005

Inadvertent Energization of All Unit 2 Pressurizer Heaters

Introduction. The inspectors reviewed a Green self-revealing finding involving the unplanned energization of all Unit 2 pressurizer heaters caused by an inadequately researched maintenance procedure.

Description. On June 1, 2005, the licensee was troubleshooting spiking in the Unit 2 pressurizer level indication using a preplanned work procedure. While in the process of replacing Alarm Relay Bistable 2LC-4627-1BN in the indication circuitry, instrumentation and control (I&C) technicians lifted electrical Lead 7 per the work procedure. Lifting this lead caused a daisy chain of power losses which caused power to be lost to Relay 63X/LC-110H in the pressurizer heater circuitry. This action in turn energized all of the backup heaters and shunted the output of the pressurizer heater hand controller station, thereby, fully energizing all proportional heaters. Lifting of Lead 7 also caused the Channel 1 high pressurizer level alarm annunciator to alarm unexpectedly. With all pressurizer heaters energized, reactor coolant system pressure rose to approximately 15 psig above normal operating pressure. In the diagnosis of the high pressurizer level annunciator, operators recognized that all pressurizer heaters were energized, took manual control, and restored pressure to normal. Additionally, I&C technicians re-landed Lead 7. During inspection of this occurrence, the inspectors discovered that the scope of the work package was inadequate, because lifting the lead had not been properly researched by system engineers or work planners causing the unexpected plant response.

Analysis. The inspectors determined that the licensee's failure to adequately research the effects of their maintenance on the pressurizer level circuitry was a performance deficiency. This finding is greater than minor because it affected the human performance attribute under the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Using the Phase 1 worksheets in MC 0609, *Significance Determination Process*, the issue was determined to have very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available. This finding had cross-cutting aspects of human performance, in that, the engineers and planners did not adequately research a procedure prior to its use on the plant.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because it occurred on nonsafety-related plant equipment. Licensee personnel entered this issue into the CAP as CR-ANO-2-2005-1678. This issue is being treated as a finding: FIN 05000368/2005004-05, "Failure to Adequately Scope the Effects of Maintenance on Pressurizer Level Instrumentation."

Entergy Response

Event Description

On June 22, 2005, with Unit 2 at approximately 100% power, Instrumentation and Control (I&C) technicians, in the process of replacing an alarm relay bistable in Channel 1 of the pressurizer level instrumentation, lifted an electrical lead that resulted in loss of power to several pressurizer instrumentation components. The loss of power resulted in the pressurizer backup heaters energizing and the proportional heaters increasing to full output due to their controller being bypassed. The loss of instrument power also resulted in a high pressurizer level alarm in the Control Room. In analyzing the alarm, operators secured both trains of backup heaters by placing their handswitches in the off position, thereby terminating the pressure transient. During this event, Reactor Coolant System (RCS) pressure increased from 2200 psia (normal operating pressure) to approximately 2216 psia, RCS temperature increased from 579 to 580.5 °F. The pressurizer spray valves, which open 40% at 2225 psia and 100% at 2240 psia, were not challenged during the event. The proportional heaters remained at full output for approximately 30 minutes, when the subject electrical lead was re-terminated by I&C personnel. Condition Report CR-ANO-2-2005-1771 was initiated to address this condition.

Safety Significance

The Technical Specifications upper limit for pressurizer pressure during normal plant operation is 2275 psia. Pressurizer pressure is normally controlled at approximately 2200 psia. This event resulted in an increase of approximately 16 psia; well within the acceptable band for system pressure.

The Pressurizer pressure control system is non-safety related. Safety-related protection against over-pressure events is provided by the Plant Protective System and the pressurizer code safety relief valves (setpoint = 2500 psia). The pressurizer pressure control system limits the pressure rise in events such as the one described above by opening spray valves when the pressure rises to 25 psia above the normal operating setpoint (valves 40% open) and 40 psia above setpoint (valves 100% open). This prevents plant trips and challenges to the safety systems. There are two spray lines and two valves, either of which could have terminated the pressure rise, preventing the need for safety-related protective action, if action had not been taken by the operators. The spray valves were not required to open during this event.

There was a small reactivity increase associated with the pressure increase which resulted in a 0.07% increase in reactor power. This increase did not threaten nuclear safety due to its small and constant value.

The safety implications of this event were minimal. There were no radiological or industrial safety implications.

Regulatory Analysis

Entergy does not contest the fact that the subject work package was inadequate and has taken corrective actions to address this issue. However, Entergy does contest the characterization of the finding as "greater than minor."

The inspectors concluded that the finding was greater than minor "since it affected the human performance attribute under the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions." Using the Phase 1 worksheet in Manual Chapter (MC) 0609, *Significance Determination Process*, the inspectors also concluded that the finding was Green (very low safety significance) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available. However, in evaluating this finding using MC 0612, Appendix B, *Issue Screening*, and MC 0612, Appendix E, *Examples of Minor Issues and Cross-Cutting Aspects*, which is a stipulated prerequisite for entering MC 0609, Entergy concluded that this finding should have been characterized as minor, and therefore should not have been documented in the inspection report. The rationale for reaching this conclusion is described below.

Using the Issue Screening flowchart in Appendix B, one quickly reaches the question, Is finding greater than minor using Appendix E? A review of Appendix E shows no relevant examples, and therefore, the answer to this question is, No. This leads to the question, Is finding greater than minor using minor questions? Evaluation of the finding with respect to the minor questions is presented below.

MC 0612, Appendix B, Section 3 (Minor Questions)

1) Could the finding be reasonably viewed as a precursor to a significant event?

   No. The Technical Specification upper limit for Pressurizer pressure in normal operations is 2275 psia. Pressure is normally controlled at approximately 2200 psia. The event resulted in an increase of about 16 psia and pressure remained well within the acceptable band. The Pressurizer pressure control system is non-safety related. Safety grade protection against over-pressure events is provided by the PPS and the pressurizer code safety relief valves. The pressurizer pressure control system limits the pressure rise in events such as the one described above by opening spray valves when the pressure rises to 25 psia above setpoint (valves 40% open) and 40 psia above setpoint (valves 100% open). This prevents challenges to the safety systems and plant trips. There are two spray lines, each containing a single spray valve that would have automatically terminated the pressure rise, preventing any challenges to credited over-pressure protection components or devices, if the operator had not taken manual control. The spray valve control circuitry was unaffected by the loss of instrument power associated with the lifted electrical lead. Therefore, this event could not reasonably be considered a precursor to a significant event.

2) If left uncorrected, would the finding become a more significant safety concern?

   As stated above, if left uncorrected, the pressurizer spray valves would have automatically terminated the pressure/power increase without challenging any safety-related protective functions.

3) Does the finding relate to a performance indicator (PI) that would have caused the PI to exceed a threshold?

   The initiating events cornerstone focuses upon operations and events that could lead to accidents if plant systems did not intervene and requires reporting of "unplanned events that result in significant changes in reactor power." The subject finding could be interpreted to relate to the initiating events cornerstone in that it resulted in an unplanned change in reactor power. However, the 0.07% power increase associated with this finding is 0.0035% of the NRC Reactor Oversight Process (ROP) reporting threshold for unplanned power changes of >20% and therefore, does not represent a "significant change in reactor power."

   NRC Inspection Procedure 61706, *Core Thermal Power Evaluation*, provides guidance that puts the subject power excursion in perspective. It states, "It is permissible to exceed the "full, steady-state licensed power level" by as much as 2% for as long as 15 minutes. In no case should 102% power be exceeded, but lesser power "excursions" for longer periods should be allowed, with the above as guidance. For example, 1% excess for 30 minutes and 0.5% for 1 hour should be allowed. Considering this guidance and the PI threshold for reporting unplanned power changes, it should be clear that the ROP does not intend that power "excursions" of the magnitude of the one described above during this event be characterized as transients and reported as unplanned power changes under this PI.

4) Is the finding associated with one of the cornerstone attributes... and <u>does the finding affect the associated cornerstone objective?</u>

   The finding does impact the human performance attribute of the initiating events cornerstone. However, it does not affect the associated cornerstone objective "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." Plant stability was not upset in that normal operating parameters were not exceeded during the event. The associated power excursion was extremely small. In fact, no safety functions would have been challenged even if the pressure/power excursion had not been terminated by the operators. With no operator action taken, the spray valves would have automatically opened when pressure reached 2025 psia, thereby terminating the pressure increase.

5) N/A

Conclusion

This event did not result in an upset in plant stability and did not challenge safety systems since plant parameters never exceeded the normal operating band and the unplanned increase in RCS pressure and reactor power would have been terminated by normal control systems even if operator action had not been taken. Therefore, this event did not impact the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability **and** challenge critical safety functions.

In summary, Entergy agrees that the unplanned changes in RCS pressure and reactor power associated with the subject finding could be interpreted as relating to the initiating events cornerstone and that it then impacted the human performance attribute of the cornerstone. However, Entergy does not agree that this finding should be characterized as "greater than minor." Considering the questions and answers above and utilizing the Issue Screening Flowchart of MC 0612, Appendix B, it can be seen that this finding should have been characterized as Minor and did not warrant evaluation under MC 0609.

Entergy requests that the NRC withdraw the Finding and the associated human performance cross-cutting aspects associated with the issue.

Non-Cited Violation 50-313/2005004-04

Excerpt from NRC Inspection Report 2005-04 for ANO issued on November 7, 2005

Unit 1 ERV Isolation

Introduction. The inspectors identified a Green NCV of 10CFR50.65(a)(4) for the failure to perform an adequate risk assessment associated with the manual isolation of the Unit 1 ERV.

Description. On August 25, 2005, Unit 1 operators noticed that the acoustic monitor indication for the Unit 1 pressurizer ERV was not operable. Operators decided to isolate the ERV by shutting its isolation Valve CV-1000 since the ERV was considered to be inoperable with its acoustic monitoring indication out of service. Discussions among operations personnel concluded that the licensee's risk management assessment program modeled both opened and closed failure modes. They reasoned that since the ERV was isolated, it could not fail to reseat and that failure mode should not be accounted for in a risk assessment. The operators also reasoned that, since the valve was inoperable because of an indication issue, the valve was available and that failure mode should not be accounted for in the risk assessment model either. As a result, the operators assumed no impact on risk would be made when isolating the ERV.

The inspectors reviewed the licensee's assessment for the existing plant conditions and concluded that the licensee had correctly used their risk management program to assess the risk with the ongoing maintenance with HPI Pump P-36A, low pressure injection Valve CV-1429, and Inverter Y-25. The inspectors then discovered in the licensee's risk assessment program that fault trees existed which showed that with the ERV isolated, the pressurizer code safety valves would be the method of preventing reactor coolant system overpressure since they would open first on any fast breaking pressure increase transient. Additionally, the inspectors learned that the probability that the pressurizer code safety valves would not close after lifting would be increased since their probability of opening increased. From this the inspectors concluded that the licensee's risk assessment was incomplete since it did not incorporate the added risk from the increased likelihood that a pressurizer code safety valve would stick open.

Analysis. The inspectors considered that the failure to account for the risk of an isolated ERV was a performance deficiency. The inspectors determined this finding was greater than minor because it related to a licensee's risk assessment which had known errors that had the potential to change the outcome of the assessment. Using Appendix K, *Maintenance Risk Assessment and Risk Management Significance Determination Process*, of MC 0609, *Significance Determination Process*, the finding was determined to have very low safety significance (Green) because the incremental increase in core damage probability was less than 2.24×10^{-8}. In this determination, the inspectors assumed Inverter Y-25 and HPI Pump P-36A were already out of service for maintenance when the ERV was isolated. Also, the inspectors used 8 hours (between 6:59 a.m. and 3:04 p.m. on August 25, 2005) as the time of the inaccurate risk assessment, which was the time when both the ERV was isolated and Green Train of the low pressure injection system was removed from service. This issue had human performance cross-cutting aspects associated with operations personnel incorrectly assuming a component had no risk significance which resulted in a non-conservative risk assessment.

Enforcement. 10 CFR 50.65(a)(4) requires, in part, that the licensee shall assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to this, the licensee did not adequately assess risk from isolating the Unit 1 pressurizer ERV. Because of the very low safety significance and because the licensee included this condition in the CAP as CR-ANO-C-2005-1257, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000313/2005004-04, "Failure to Adequately Assess Risk for an Isolated Pressurizer Electromatic Relief Valve."

Entergy Response

Event Description

On August 24, 2005, the Unit 1 Control Room received alarm K09-A1, *Relief Valve Open*, while taking logs on VYI-1000A, Electromatic Relief Valve (ERV) monitor. Utilizing procedure OP-1105.013, *Pressurizer Relief Valve Monitoring System*, Control Room Operators attempted to switch channels; however, the monitor remained pegged high. Abnormal Operating Procedure (AOP) OP-1203.015, *Pressurizer System Failure*, was then entered which directed the Operators to close the ERV isolation valve, CV-1000, and to maintain the ERV vent path closed. The AOP also required the addition of a caution tag on the CV-1000 hand switch providing guidance for use of the valve for Emergency Operating Procedure (EOP) purposes.

Operations personnel were keenly aware of the significance of the ERV and the impact of closing the isolation valve. The impact of isolating the ERV and the contingency steps required to be performed to support future use of the ERV were discussed with the Control Room staff and the Assistant Operations Manager. An informal qualitative risk assessment, which considered ongoing Inverter Y-25 and High Pressure Injection (HPI) pump P-36A maintenance, determined that the closure of the ERV isolation valve had an insignificant impact to plant risk. Condition report CR-ANO-1-2005-01238 was generated and work request 59334 was initiated to ensure timely restoration.

During the next shift, "B" decay heat cooler outlet valve, CV-1429, was removed from service for testing. At that time, "B" low pressure injection train was declared inoperable and the plant entered an "acceptable risk" category. The risk assessment considered the previously discussed on-going maintenance and the isolated ERV.

Safety Significance

The risk associated with isolating the ERV was appropriately considered. Although the ERV was isolated, it remained available should it have been required for RCS pressure control or core cooling. The Control Room staff was made aware of the compensatory steps required to operate the ERV isolation valve.

The safety significance of this condition was minimal as there was no required change to risk classification nor were there any risk management actions.

Regulatory Analysis

The inspectors concluded that the finding was greater than minor "because it related to a licensee's risk assessment which had known errors and had the potential to change the outcome of the assessment. Using Appendix K, *Maintenance Risk Assessment and Risk*

Management Significance Determination Process, of MC 0609, *Significance Determination Process*, the finding was determined to have very low safety significance (Green) because the incremental increase in core damage probability was less than 2.24 X 10^{-8}."

However, in evaluating this finding using MC 0612, *Power Reactor Inspection Reports*, the documentation process begins with the screening of inspection results to determine if an inspection issue must be documented in an inspection report. The steps to screen inspection results are described in Section 0612-05. First, the performance deficiency question in Appendix B, Section 1 of MC 0612 must be considered.

> Did the licensee fail to meet a requirement or standard, where the cause was reasonably within the licensee's ability to foresee and correct and which should have been prevented?

As discussed above, control room personnel were aware of the significance of the ERV and of closing the isolation valve. An informal qualitative risk assessment was performed that determined that the closure of the ERV isolation valve had an insignificant impact to plant risk; therefore, there was no failure to meet a requirement or standard. As outlined in Section 0612-05, since the answer to the performance deficiency question is "No", then the issue is not a performance deficiency.

Per NUMARC 93-01, Revision 3, Section 11.3.2, the assessment method may use quantitative approaches, qualitative approaches, or blended methods. When using the qualitative approach, the risk assessment should consider, "the degree of redundancy available for performance of the safety function(s) served by the out-of-service SSC" and, "the likelihood that the maintenance activity will significantly increase the frequency of a risk significant initiating event (e.g., by an order of magnitude or more as described by each licensee, consistent with its obligation to manage maintenance-related risk)." These elements of our qualitative risk assessment were performed satisfactorily. In addition, procedures were used which had previously incorporated risk insights that required the closing of the block valve if leak detection is inoperable in accordance with previous Technical Specification requirements and as submitted in our response to Generic Letter 90-06. Therefore, there was no failure to meet a requirement or standard.

Conclusion

This issue was not a performance deficiency; therefore, Section 05-02 of MC 0612 should not have been entered and the issue should not have been screened as "Greater than Minor." Entergy does not agree that a finding occurred and requests that the NRC withdraw the NCV and the associated human performance cross-cutting aspects associated with the issue.

Apparent Violation (From IR 05000313/368/2005003 Closed in IR 05000313/368/2005004)

Excerpt from NRC Inspection Report 2005-04 for ANO issued on November 7, 2005

Inadequate Procedure Leads To Reactor Coolant Pump Seal Damage

Introduction. The inspectors completed the significance determination of the apparent violation documented in NRC Inspection Report 05000313/2005003 and 05000368/2005003. The apparent violation involved an inadequate procedure related to the alignment of reactor coolant pump (RCP) seal injection flow when the pump and motor were uncoupled. An additional entry into reduced reactor coolant system (RCS) inventory conditions during the refueling outage was necessary to repair the damaged RCP seal caused by this performance deficiency.

Description. During Unit 2 Refueling Outage 2R17, on March 13, 2005, operators commenced filling of the RCS after a period of reduced inventory to install steam generator nozzle dams. Section 8.0, *RCS Fill Operations*, of Procedure 2103.002, *Filling and Venting the RCS*, Revision 39, instructed operators to align seal injection to all RCPs as part of the fill evolution. In their efforts to align seal injection to RCP 2P-32C, operators encountered difficulties attaining adequate flow, so they adjusted seal flow but observed abnormal seal pressures. The licensee had replaced the motor for RCP 2P-32C earlier in the outage and the pump and motor for RCP 2P-32C were still uncoupled. The pump and motor should have been recoupled prior to initiating seal injection flow to the pump. The observed abnormal pressures and difficulty in establishing seal injection flow were captured in the licensee's corrective action program in CR ANO-2-2005-0545. In the operability evaluation for this CR, engineers declared the seal operable. Their determination was, in part, based on a discussion with the seal vendor. However, in this conversation, the engineers did not make it clear to the vendor that the pump was uncoupled during the periods of observed abnormal pressure. Replacement of the seal would have been desired at this time, since the overall risk of the outage would have been minimized because the plant was defueled at this time.

The damaged seal went undetected until April 4, 2005, when operators commenced filling the RCS in preparation for returning to power operations. Initial RCS level was 84 inches, which was just below the reactor vessel flange level. At approximately 188 inches in the RCS, operators noticed an estimated 26 gallon per minute leak from the RCP seal and secured the RCS fill activity. The RCS was subsequently drained to the 90-inch level.

The licensee entered reduced RCS inventory conditions at 11:01 p.m. on April 4, 2005, and continued draining to seal replacement level. They remained in reduced inventory to replace the seal until 5:12 a.m. on April 6, 2005, (approximately 30 hours). During this reduced inventory activity, RCS temperatures were controlled between 112°F and 129°F and time-to-boil was approximately 1 hour. The inspectors considered this an unplanned entry into reduced RCS inventory. Additionally, the inspectors considered the lower inventory of the RCS to be an affected mitigating system for the prevention of boiling conditions in the reactor vessel. The seal replacement that resulted in the unplanned reduced inventory condition was successfully completed.

Analysis. The inspectors considered that the failure to have an adequate procedure for ensuring isolation of seal injection when a Unit 2 reactor coolant pump was uncoupled was

a performance deficiency. Traditional enforcement does not apply for this finding because it did not have any actual safety consequences or potential for impacting the NRC's ability to perform its regulatory function nor was it the result of any willful violation of NRC requirements. The inspectors determined that this finding is greater than minor because it was associated with the mitigating systems cornerstone configuration control attribute and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors used Appendix G, *Shutdown Operations Significance Determination Process*, of MC 0609, *Significance Determination Process*, to further determine the significance of this finding.

Unplanned entry into reduced RCS inventory conditions to repair the RCP seal represented additional risk incurred above the planned outage risk. The additional risk associated with the reduced RCS inventory evolution constitutes the additional risk incurred above the planned outage risk. A Phase 1 screening of the finding was performed using Appendix G and the Attachment 1 checklists. The finding was not considered a "Loss of Control" using Table 1. Using Checklist 3, "PWR Cold Shutdown and Refueling Operation - RCS Open and Refueling Cavity Level < 23' Or RCS Closed and No Inventory in Pressurizer, Time to Boiling < 2 hours," in Attachment 1, "Phase 1 Operational Checklists for both PWRs and BWRs," of Appendix G of MC 0609, the inspectors determined this finding required quantitative assessment because the finding increased the likelihood of a loss of RCS inventory by requiring an additional entry into a reduced RCS inventory condition. Therefore, the finding was referred to the regional senior reactor analyst for further evaluation.

Since the finding did not involve low temperature overpressure protection, nozzle dams, or boron dilution, the analyst used Appendix G, Attachment 2, "Phase 2 SDP Template for PWR During Shutdown." The finding involved an additional entry into a high-risk Plant Operating State (POS). Therefore, as cautioned in Attachment 2, the senior reactor analyst consulted with staff in the Office of Nuclear Reactor Regulation to evaluate the change in core damage frequency associated with the finding. The following is a summary of the analysis that was performed.

Result. The risk significance of the finding from this point is determined in the same manner as for at-power findings. Using MC 0609, Appendix A, Step 2.4, "Estimating the Risk Significance of Inspection Findings," the analyst summed the quantified sequences and determined that the total increase in core damage frequency associated with this finding due to internal initiating events was estimated as 1 E-7/year using the counting rule. No screening for potential contribution due to external events or large early release frequency was performed because of the assumed conservative upper-bound screening result provided by the SDP worksheets. Therefore, this was a finding of very low safety significance (Green). Contributing to this result was that (1) the seal replacement activity required RCS draindown to reduced inventory conditions and not to midloop conditions, (2) the time needed to replace the seal was not extensive and, (3) the time after shutdown provided additional time available for successful operator actions.

Enforcement. The inspectors determined that since Procedure 2103.002, "Filling and Venting the Reactor Coolant System," Revision 39, was inadequate, it did not meet the requirements of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, and as a

result the licensee did not meet Unit 2 Technical Specification 6.4.1, "Procedures." Because of the very low safety significance of this finding and because the licensee included this condition in their CAP as CR ANO-2-2005-0545, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000368/2005004-07, "Inadequate Procedure Leads To Reactor Coolant Pump Seal Damage."

It was also noted that the cause of the finding is related to the cross-cutting element of human performance.

Entergy Response

Event Description

On March 12, 2005, Work Order (WO) 00034757-05 was implemented for mechanical support of the 2P-32C motor replacement project. After the tags were installed, the pump was uncoupled by locking the 2P-32C seal shaft sleeve in place using the locking pins, lowering the pump shaft to rest on the recirculation impeller, and removing the coupling spool piece. This procedure contained no notes or cautions with respect to prohibiting the use of seal injection while in the uncoupled state.

By lowering the pump shaft to rest on the recirculation impeller, communication between the cavity immediately below the seal package (where seal injection flow enters) and the RCS was blocked. When seal injection is applied in this configuration, the seal cavity is pressurized.

While the pump was uncoupled, Operations was in the process of raising RCS level. Operations procedure OP-2103.002, *Filling and Venting the RCS*, requires the operators to establish seal injection when raising RCS level. The purpose of utilizing seal injection in this configuration is to ensure only clean RCS fluid from the Chemical Volume Control and System (CVCS) is supplied to the seal cavity while the RCS level is raised. This procedure contained no notes or cautions with respect to prohibiting the use of seal injection when a pump is uncoupled.

The N-9000 seal has a 9" balance diameter, and the o-rings at the upper and lower ends of the shaft sleeve seal have a balance diameter of 7.25" and 7.625" respectively, which yields a hydraulic thrust area of approximately 22.9 square inches. Seal injection at >900 psi would result in an ~20,000 pound up-thrust on the shaft sleeve. During normal operation, the thrust is transmitted through the adjusting ring to the pump half coupling. However, in this situation, the spool piece had been removed and the shaft sleeve locating pins were installed at the time seal injection pressure was established to the seal. Without the spool piece in place, the up-thrust reacted against the locating pins. The locating pins are not designed to handle the upward movement caused by the large loads that were introduced.

Consequently, the upward thrust of the shaft sleeve caused the pins to bend and the stationary faces to be pushed to their upper axial limits. Crud buildup on the balance sleeve of the seasoned seal caused the stationary face and subassemblies to bind. Therefore, the faces of the seal were open when Maintenance personnel lowered the pump assembly to reset the running position of the shaft sleeve. With the seal faces open, excessive leakage migrated through the seal and leaked out between the fourth stage and

the seal adjusting ring. Based upon this leakage, the decision was made to replace the seal. The N-9000 seal was partially disassembled to validate the cause of leakage.

Root Cause

Entergy determined the root cause of this condition to be the lack of vendor information and Operating Experience (OE). Because of this lack of information, it was not understood that seal damage could occur by pressurizing the seal with the pump uncoupled. Plant procedures associated with maintenance of an RCP seal and with filling and venting the RCS lacked important information that needed to be in place to reduce the potential for inadvertently damaging plant equipment.

A search of previous ANO corrective action documents and industry operating experience was performed. There were no other documented instances of seal failures that resulted from seal injection pressure being introduced while a pump was uncoupled. Vendor documents were also reviewed and although several seal-related documents were found, none were identified that addressed issues and concerns with operating seal injection while a pump is uncoupled.

Safety Significance

The health and safety of the general public, nuclear safety, and radiological safety was not jeopardized by this condition. During normal operation, the seal is designed to absorb this type of movement when thrust in the upward direction. There was no mechanical damage to the seal as a result of this event. However, due to its age and the crud deposits on the balancing sleeve, the gross movement created by the pressurization caused the stationary faces to be moved up on the balancing sleeve and the seal faces to remain in the up position. This condition could not have been initiated during any operating scenario as the position of the shaft sleeve is governed by the location of the pump rotating assembly and seal integrity is verified prior to plant startup. The potential does exist for an inadvertent loss of inventory while in the refuel process. However, the inventory loss can be corrected by reducing RCS level and repairing the seal as was accomplished in the case discussed above.

The reduced inventory maintenance was carried out without incident. Decay heat removal was carried out in accordance with procedural guidance and training. No equipment malfunction that impacted decay heat removal operation occurred. Inventory was controlled in accordance with procedures and training. No loss of inventory control was experienced. Required power was available throughout the maintenance evolution and containment closure guidance was adhered to.

Therefore, there is little or no safety significance associated with this condition.

Regulatory Analysis

Per Appendix B to MC 0612, for an issue to be considered a finding it must first be determined if the issue is a performance deficiency. To determine if an issue is a performance deficiency, the following question must be answered positively:

> Did the licensee fail to meet a requirement or a standard, where the cause was reasonably within the licensee's ability to foresee and correct and which should have been prevented?

As determined in the ANO root cause, Entergy had no foreknowledge of the need for procedural controls to prohibit applying seal injection to an uncoupled RCP seal nor was there any OE data identifying this need. Seal injection is procedurally required during RCS fills to ensure only pure RCS fluid is applied to the seal cavities. The operators were following plant procedures during this evolution. Vendor documentation provided no cautions to make licensees aware of this potential condition. To understand the need for procedural cautions to guard against this condition, one would have to possess both a detailed knowledge of the seal design and an understanding that there may be an occasion where seal injection would be utilized on an uncoupled pump. Therefore, it was not reasonable for Entergy to have foreseen the need to have procedural guidance that would have prohibited applying seal injection to an uncoupled RCP seal or to have perceived the need to add these limitations to the work package associated with the RCP motor replacement.

In light of the above information, this issue does not constitute a finding. These facts also do not support the characterization of this issue as having human performance cross-cutting aspects, as noted in the inspection report.

Additionally, even if this issue were to be characterized as a finding, Entergy does not agree that its significance should be considered any greater than minor. Given there are no examples similar to this condition in Appendix E of MC 0612, the minor questions in Appendix B must be used to determine the significance of the finding. The inspector determined that only question 4 applied.

> (4) Is the finding associated with one of the cornerstone attributes listed at the end of this attachment and <u>does the finding affect the associated cornerstone objective</u>?

With regard to this question, the inspector determined that this finding is greater than minor because it was associated with the mitigating systems cornerstone configuration control attribute and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating event to prevent undesirable consequences.

Entergy does not agree that a maintenance condition that requires a reduced inventory window to correct, would <u>affect the mitigating system cornerstone objective</u> of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The maintenance activity did not result in any impact on mitigating system performance. Systems required to ensure core heat removal, inventory control, power availability, and containment closure where fully operational, with appropriate procedural guidance and operators trained on their use. Therefore, the finding did not affect the associated cornerstone objective and should not be characterized as greater than minor, assuming the issue is in fact considered a finding (see above discussion).

Conclusion

Entergy does not agree with the NRC's conclusion that this issue constitutes a performance deficiency. Entergy had no foreknowledge of the need for procedural controls to prohibit applying seal injection to an uncoupled RCP. Without detailed knowledge of the seal

design, this limitation is not readily apparent. No regulation or standard was violated. Because this condition is not addressed in vendor guidance or operating experience, the lack of procedural guidance should not be characterized as having human performance cross-cutting aspects, as noted in the inspection report.

Furthermore, if the issue were considered a performance deficiency it should not be considered more than minor given the mitigating system cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was not affected. The maintenance activity did not result in any impact on mitigating system performance. Systems required to ensure core heat removal, inventory control, power availability, and containment closure were fully operational with appropriate procedural guidance and operators trained on their use.

Entergy does not agree that a finding occurred and requests that the NRC withdraw the NCV and the associated human performance cross-cutting aspects associated with the issue.