UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

February 28, 2006

Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, Louisiana 70775

SUBJECT: RIVER BEND STATION - NRC PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION REPORT 0500458/2005008

Dear Mr. Hinnenkamp:

On December 8, 2005, the Nuclear Regulatory Commission (NRC) completed the onsite portion of a team inspection at your River Bend Station. The enclosed report presents the results of the inspection, which were discussed on December 8, 2005, with Mr. R. King and other members of your staff. The team continued in-office document reviews and conducted a final exit meeting with Mr. R. King and other members of your staff on January 19, 2006.

This inspection examined activities conducted under your license as they relate to the identification and resolution of problems, compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed approximately 225 condition reports, apparent cause and root cause analyses, as well as supporting documents. In addition, the team reviewed cross-cutting aspects of NRC and licensee-identified findings and interviewed personnel regarding the safety conscious work environment.

On the basis of the sample selected for review, the team concluded that, in general, your processes to identify, prioritize, evaluate, and correct problems were effective; thresholds for identifying issues remained appropriately low and, in most cases, corrective actions were adequate to address conditions adverse to quality. Notwithstanding the above, poor engineering rigor associated with evaluating and prioritizing issues resulted in a relatively high number of self-revealing and NRC identified findings at your site. Some of these findings culminated in plant scrams and/or complicated operator response to emergency events. Others were related to equipment deficiencies, some of which resulted in inoperable safety-related equipment. The team concluded that a positive safety-conscious work environment existed at your River Bend Station.

The report documents four findings that were evaluated under the risk significance determination process as having very low safety significance (Green). All of the findings were associated with violations of NRC requirements. The violations are being treated as non-cited violations because they are of very low safety significance and because they have been entered into your corrective action program consistent with Section VI.A of the Enforcement Policy. If you contest the violations or the significance of these non-cited violations, you should provide a response within 30 days of the date of the inspection report, with the basis for your denial, to the U.S. Nuclear Regulator Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

//RA//

Linda Joy Smith, Chief
Engineering Branch 2
Division of Reactor Safety

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Licenses: NPF-47

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-458
License:	NPF-47
Report:	05000458/2005008
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 US Highway 61 St. Francisville, Louisiana
Dates:	November 14, 2005 through January 19, 2006
Inspectors:	Z. Dunham, Senior Resident Inspector, Branch A, DRP G. Replogle, Senior Reactor Inspector, Engineering Branch 1, DRS M. Miller, Resident Inspector, Branch C, DRP J. Drake, Operations Engineer, Operations Branch, DRS
Approved by:	L. J. Smith, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000458/2005008; 11/14/2005 - 1/19/2006; River Bend Station; biennial baseline inspection of the identification and resolution of problems. Violations were identified in the areas of prioritization and evaluation of issues.

The inspection was conducted by a senior resident inspector, a senior reactor inspector, a resident inspector, and an operations engineer. Four Green findings classified as non-cited violations of very low safety significance were identified during this inspection. The findings were evaluated using the significance determination process. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

<u>Identification and Resolution of Problems</u>

- The team reviewed approximately 225 condition reports, apparent and root cause analyses, as well as supporting documents to assess problem identification and resolution activities. In general, the corrective action program procedures and processes were effective, thresholds for identifying issues were low, and corrective actions were adequate to address conditions adverse to quality. Notwithstanding the above, poor engineering rigor associated with the prioritization and evaluation of issues resulted in a relatively high number of self-revealing and NRC identified findings. Some of these findings culminated in plant scrams and/or complicated operator response to emergency events. Others were related to equipment deficiencies, some of which resulted in inoperable safety-related equipment.

  Based on the interviews conducted, the team concluded that a positive safety conscious work environment exists at River Bend Station. The team determined that employees felt free to raise safety concerns to station managers and supervisors, the employee concerns program, and the NRC. However, the team received a few isolated comments regarding the correction action program feedback process. These individuals had previously identified corrective action issues and were not satisfied with the program's responses to their concerns. Some of these individuals commented that they were hesitant to use the corrective action program in the future. The licensee acknowledged the comments and planned to take action to address the concerns. All the interviewees believed that potential safety issues were being addressed.

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Green</u>. The team identified a non-cited violation of Technical Specification 5.4.1.a (Procedures) for unacceptable preconditioning of a low pressure core spray keepfill system check valve. The test procedure failed to prescribe testing the check valve in the as-found condition. Instead (during testing of the system pump) the document directed operators to flush the valve at 27 gpm for up to 20 minutes prior to the check valve test. Corrosion buildup in the valve, which had previously caused valve failures, was a known concern and the preconditioning could have masked performance problems. Failure of the valve to perform its safety function puts the low pressure core spray system at risk of water hammer during a loss of offsite power event. The licensee planned to test the valve in the as-found configuration during future tests. The licensee documented this issue in their corrective action program as CR-RBS-2005-04123.

The failure to properly test the subject check valve was a performance deficiency. The finding was more than minor because, if left uncorrected, the problem could result in a more significant safety concern. Specifically, the surveillance test may not identify valve failure. The finding was of very low risk significance because it was not a design/qualification issue, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its technical specification allowable outage time, did not result in a loss of function of non safety-related risk significant equipment and was not risk significant due to external events. The finding had problem identification and resolution cross-cutting aspects because the licensee had failed to properly evaluate the issue as preconditioning in response to readily available industry information (Section 4OA2.e.(2)(I)).

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI (Corrective Actions) for the failure to take prompt corrective measures to address a significant condition adverse to quality. Specifically, the low pressure core spray keepfill pump discharge check valve failed on two occasions (significant conditions adverse to quality) and planned corrective measures to replace the check valve were not timely. The check valve failures put the low pressure core spray system at increased water hammer risk during a loss of offsite power event. The licensee had identified that corrosion buildup was causing the valve to leak excessively when closed. The licensee documented this issue in their corrective action program as CR-RBS-2005-04162 and planned to replace the valve at the next available opportunity.

  The failure to take prompt corrective measures to address a significant condition adverse to quality was a performance deficiency. The finding was greater than minor because it was an equipment performance reliability issue which impacted the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events. Using Manual Chapter 0609, “Significance Determination Process,” Phase 1 worksheet, the finding was of very low risk significance because it was not a design/qualification issue, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its technical specification allowable outage time, did not result in a loss of function of non safety-related risk significant equipment and was not risk significant due to external events. The finding had cross-cutting aspects in the area of problem identification and resolution (Section 4OA2.e.(2)(ii)).

- Green. The team identified a 10 CFR 50, Appendix B, Criterion V (procedures) non-cited violation for the failure to set safety-related limit switches in accordance documents appropriate to the circumstances for 34 safety-related throttle valves. The licensee set motor-operated valve (MOV) open indication light limit switches so that the open indication de-energized between the 95% and 100% closed positions, whereas the applicable procedure and design drawing required that the limit switches be set to the 100% closed position. This practice had caused repetitive operational problems in the plant. The licensee entered this issue into their corrective action program as CR-RBS-2005-04113.

  The failure to adjust MOV limit switches in accordance with documents appropriate to the circumstances was a performance deficiency. The issue was more than minor because it affected the mitigating systems cornerstone objective, in that it affected the operability, availability, reliability or function of a system or train in a mitigating system. The finding was of very low safety significance because it was a design/qualification deficiency confirmed not to result in loss of operability per "Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment." This finding had cross-cutting aspects in the areas of human performance, (the failure to follow procedures) and problem identification and resolution because the licensee failed to identify the problem in response to a prior related NRC violation (Section 4OA2.e.(2)(iv)).

Cornerstone: Barrier Integrity

- Green. The team identified two examples of a Technical Specification 3.2.2, "Minimum Critical Power Ratio" (MCPR), non-cited violation for the failure to prevent transition boiling on the fuel during Operational Cycles 8 and 11. Fuel failures due to transition boiling were experienced during each cycle. Engineers failed to properly understand the affect of zinc injection on the cladding surfaces following the Cycle 8 fuel pin failures and zinc injection was reinitiated before the corrective actions to prevent recurrence were in place. The licensee had industry information that indicated that zinc injection contributed to the accumulation of loose crud and the formation of tenacious crud on the fuel. The additional crud rendered the Technical Specifications Minimum Critical Power Ratio (MCPR) calculations inaccurate and transition boiling occurred in localized areas. The licensee entered this issue into their corrective action program as CR-RBS-2006-0255.

  The failure to prevent transition boiling in the core was a performance deficiency. The issue was more than minor because it impacted the barrier integrity cornerstone objective to maintain the integrity of the fuel cladding. The finding screened out as of very low safety significance (Green) because it only affected the fuel barrier. The issue had cross-cutting aspects in the areas of problem identification and resolution, in that the licensee failed to properly evaluate pertinent related industry information, which could have precluded the first violation, and failed to properly implement effective corrective measures in response to the first set of fuel failures, which led to the second violation (Section 4OA2.e.(2)(iii)).

B. Licensee-Identified Violations

None.

REPORT DETAILS

4 OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

The team based the following assessments, in part, on issues that were identified in the assessment period, which ranged from November, 2003 (the last biennial problem identification and resolution inspection) to the end of the onsite portion of the inspection on December 8, 2005. The referenced issues came from all inspection efforts conducted during the period. The examples are divided into two groups. The first group (Current Issues) includes problems that were identified during the assessment period where the performance concern also occurred during the same period. The second group (Historical Issues) includes issues that were identified during the assessment period but all the performance deficiencies occurred outside the period of interest.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The team reviewed items selected across the seven cornerstones to determine if problems were being properly identified, characterized, and entered into the corrective action program. The team performed equipment walkdowns, and reviewed operator logs, maintenance records, and equipment deficiency tracking logs for equipment deficiencies that should have also been captured in the corrective action program. In addition, the team reviewed a sample of licensee audits and self assessments, trending reports, system health reports, and various other reports and documents related to the corrective action activities.

The team interviewed station personnel and evaluated corrective action documentation to determine the licensee's threshold for identifying problems. In addition, in order to assess the licensee's handing of operator experience, the team reviewed the licensee's evaluation of selected industry operating experience reports, including licensee event reports, NRC Generic Letters, Bulletins and Information Notices, and generic vendor notifications.

(2) Assessment

The team determined that, in general, problems were properly identified and entered into the corrective action program as evidenced by the relatively few findings identified during the assessment period. However, the licensee did fail in some instances to identify or document deficiencies which directly contributed to one plant scram and complicated recovery following a second scram.

Current Issues

Example 1: The licensee failed to identify a deficient surveillance practice at the first opportunity. Specifically, technicians were preconditioning (cycling) breakers prior to speed testing. The preconditioning masked degraded breaker performance and on August 15, 2004, slow breaker operations in response to a ground fault caused multiple unnecessary breaker trips, a plant trip and a partial loss of offsite power (self-revealing, NRC Inspection Report 05000458/2004005).

Example 2: On August 15, 2004, operators had discovered, but failed to properly document in the corrective action program or plant procedures, certain manual actions that were necessary to maintain condenser vacuum in response to a partial loss of offsite power. Consequently, in response to a similar event, main condenser vacuum degraded and unnecessarily complicated operator response to a plant scram (self-revealing, NRC Inspection Report 05000458/2004012).

Example 3: The NRC identified that the licensee missed two opportunities to identify design control issues associated with tape covering the louvers on top of auxiliary building 480 Vac engineered safety features Switchgear EJS-SWGR2A (NRC Inspection Report 05000458/2004003).

Example 4: The NRC identified that the licensee missed a prior opportunity to correct a lack of understanding by operators on the proper operation of motor-operated throttle valves. As a result, an operator failed to fully close a safety-related valve and water was inadvertently transferred from the suppression pool to the upper pool (NRC Inspection Report 05000458/2005004).

Historical Issues

Example 5: The licensee missed several opportunities to identify inaccurate feedwater flow measurement instrumentation. Consequently, the reactor exceeded the licensed maximum power level from February 27, 1996 to May 10, 2003 (self-revealing, NRC Inspection Report 05000458/2004002).

Example 6: The NRC identified that, in 1993, the licensee failed to perform vendor recommended magnetic particle inspections on two emergency diesel generator cylinder liners. This inspection was a corrective action in response to a 10 CFR 21 report (NRC Inspection Report 05000458/2005004)

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The team reviewed condition reports and operability evaluations to assess the licensee's ability to evaluate adverse conditions. The team reviewed a sample of condition reports, apparent cause analyses and root cause analyses to ascertain whether the licensee properly considered the full extent of conditions, generic implications, common causes, and previous occurrences.

In addition, the team reviewed licensee evaluations of selected industry operating experience reports, including licensee event reports, NRC Generic Letters, Bulletins and

Information Notices, and generic vendor notifications to assess whether issues applicable to River Bend Station were appropriately addressed.

The team performed a historical review of condition reports written over the last five years that addressed Class 1E 4160 VAC and 480 VAC breakers, the reactor core isolation cooling system, the high pressure core spray system, emergency diesel generators, standby service water, and 125 VDC batteries.

(2) Assessment

The team concluded that problems were generally prioritized and evaluated in accordance with the licensee's corrective action program guidance and NRC requirements. The team found that for the sample of root cause reports reviewed, the licensee was generally self-critical and thorough in evaluating the causes of significant conditions adverse to quality. Notwithstanding the above, poor engineering rigor associated with prioritization and evaluation of issues resulted in a relatively high number of self-revealing and NRC identified findings. Some of these findings culminated in plant scrams and/or complicated operator response to emergency events. Others were related to equipment deficiencies, some of which resulted in inoperable safety-related equipment.

Current Issues

Example 1: The licensee failed to properly evaluate three prior occurrences of slow to open switchyard breakers. Consequently, on August 15, 2004, slow breaker operations in response to a ground fault caused multiple unnecessary breaker trips, a plant trip and a partial loss of offsite power (self-revealing, NRC Inspection Report 05000458/2004005).

Example 2: The licensee failed to promptly address degraded circulating water cooling tower drift eliminators. Consequently, contamination accumulated on transformer yard insulators and caused a ground fault, a reactor scram and a partial loss of offsite power (self-revealing, NRC Inspection Report 05000458/2004012).

Example 3: Engineers failed to adequately address repetitive auxiliary building roof leaks. After several instances of roof leaks, on February 5, 2004, rainwater inleakage through the roof resulted in an electrical ground on the safety-related 480 VAC switchgear control circuits (self-revealing, NRC Inspection Report 05000458/2004002).

Example 4: The NRC identified that the licensee failed to properly assess a failure of the station blackout diesel generator starting system. This resulted in an additional 24 hours of diesel unavailability (NRC Inspection Report 05000458/2005004).

Example 5: The NRC identified that the licensee failed to take prompt corrective measures to address a condition adverse to fire protection. The licensee relied on compensatory measures for seven years instead of correcting a fire protection coating deficiency in three areas important to safe shutdown (NRC Inspection Report 05000458/2004007).

Example 6: The licensee failed to take prompt corrective measures following a February 5, 2004, electrical ground caused by auxiliary building roof leaks. Consequently, on December 5, 2004, roof leaks caused a loss of auxiliary building area unit Cooler HVC-UC11A (self-revealing, NRC Inspection Report 05000458/2004005).

Example 7: The NRC identified that the licensee failed to take prompt and effective corrective measures in response to reports of missed portable fire extinguisher inspections. 28 condition reports documented missed portable fire extinguisher inspections between January 2000 and April 2005 (NRC Inspection Report 05000458/2005003).

Example 8: The licensee identified that the site failed to take prompt corrective measures to address a Division 1 emergency diesel generator jacket water system fitting leak. Consequently, the fitting failed and rendered the diesel generator inoperable (NRC Inspection Report 05000458/2005004).

Example 9: The NRC identified that the engineers, while evaluating a degraded low pressure core spray system check valve, failed to identify preconditioning of the valve during testing. Industry information on preconditioning was readily available (see Section 4OA2.e(2)(I) of this report).

Example 10: The NRC identified that the licensee failed to promptly correct a degraded low pressure core spray keepfill pump discharge check valve (NRC Identified, see Section 4OA2.e(2)(ii) of this report).

Example 11: The NRC identified that plant engineers failed to properly evaluate a prior violation. Following an instance where an operator failed to fully close a motor-operated valve (MOV) due to misleading valve position indication, plant engineers failed to identify that the valve indication lights were not set in accordance with design requirements (see Section 4OA2.e(2)(iv) of this report).

Historical Issues

Example 12: The licensee failed to properly evaluate the cause of a cracked turbine control system hydraulic line which caused a reactor scram on August 31, 2000. Another plant scram occurred on February 22, 2003, when a similar turbine control system hydraulic line also failed. (self-revealing, NRC Inspection Report 05000458/2003006).

Example 13: The NRC identified that the licensee failed to identify the root cause of a April 21, 2001, turbine trip and reactor scram. This event was caused by electrostatic arcing that affected the primary and backup turbine speed probes. Consequently, a similar event occurred on September 22, 2003 (NRC Inspection Report 05000458/2004005).

Example 14: The NRC identified that the engineers failed to fully understand the affects of zinc injection and the potential impact on minimum critical power ration calculations. Consequently, zinc injection, in conjunction with high feedwater iron levels, led to fuel failures during two operating cycles. While engineers failed to understand the root cause of the first set of fuel failures, zinc injection was again initiated, which led to the second recurring set of fuel failures (see Section 4OA2.e(2)(iii) of this report).

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The team reviewed plant records, primarily condition reports, to verify that corrective actions related to the issues were identified and implemented, including corrective actions to address common cause or generic concerns. The team sampled specific technical issues to evaluate the adequacy of the licensee's operability determinations.

Finally, the team reviewed a sample of condition reports that addressed past NRC identified violations, for each affected cornerstone, to ensure that the corrective actions adequately addressed the issues as described in the inspection reports. The team also reviewed a sample of corrective actions closed to other condition reports or work process documents to ensure that corrective actions were still appropriate and timely.

(2) Assessment

The effectiveness of identified corrective actions to address adverse conditions was generally adequate. The NRC identified a few instances where historical corrective actions were not effective but, overall, the licensee demonstrated acceptable performance in this area.

Current Issues

None.

Historical Issues

Example 1: The licensee failed to implement previously identified corrective actions associated with minimizing the likelihood of making electrical contact on adjacent relays and circuits while installing electrical jumpers. As a result, a loss of power to Division I ESF switchgear and an inadvertent start of the Division I emergency diesel generator occurred (self-revealing, NRC Inspection Report 05000458/2004005).

Example 2: The NRC identified that, in June 1997, the licensee identified that certain actions associated with emergency diesel generator surveillance testing constituted preconditioning but failed to change test procedures to correct the issues (NRC Inspection Report 05000458/2004003).

d. Assessment of Safety-Conscious Work Environment

(1) Inspection Scope

The team interviewed 27 individuals from different departments representing a cross section of functional organizations and supervisory and non-supervisory personnel. These interviews assessed whether conditions existed that would challenge the establishment of a safety-conscious work environment.

(2) Assessment

The team concluded that a safety-conscious work environment exists at River Bend Station. Based on interviews, station personnel felt free to enter issues into the corrective action program, raise safety concerns with their supervision, to the employees concern program, and to the NRC. However, the team received a few isolated comments regarding dissatisfaction with the Corrective Action Program feedback process. Corrective Action Program engineers provide feedback regarding the final dispositioning of reported issues to the originators of the concerns. In some cases, individuals were informed that their concerns were closed to other similar condition reports or were closed to "Trend." In the case where issues were closed to other conditions reports, individuals were not informed about the ultimate corrective measures to address the concerns. When concerns were closed to Trend, individuals did not always understand that hardware problems were still corrected and that the Trend designator simply meant that engineers were continuing to track similar types of problems for trending purposes. Some of these individuals commented that they were hesitant to use the corrective action program in the future given the lack of feedback. Notwithstanding the above, all the interviewees believed that potential safety issues were being properly addressed. There were no instances identified where individuals had experienced adverse consequences for bringing safety issues to the NRC. The licensee acknowledged the feedback related concerns and stated that they would take action to address the negative perceptions.

e. Specific Issues Identified During This Inspection

(1) Inspection Scope

During the reviews described in Sections 4OA2 a.(1), 4OA2 b.(1), 4OA2 c.(1), and 4OA2 d.(1), the team identified the following findings.

(2) Findings

I. Preconditioning of a Low Pressure Core Spray System Check Valve

Introduction. The team identified a Green non-cited violation of Technical Specification 5.4.1.a (Procedures) for unacceptable preconditioning of a low pressure core spray keepfill pump discharge check valve. Procedure STP-205-6301, "LPCS [Low Pressure Core Spray] Quarterly Pump and Valve Operability Test," failed to prescribe testing the check valve in the as-found condition. Instead (during testing of the system pump) the document directed operators to flush the valve at 27 gpm for up to 20 minutes prior to the check valve test. Corrosion buildup in the valve, which had previously caused valve failures, was a known concern and the preconditioning could have masked performance problems. Failure of the valve to perform its safety function puts the low pressure core spray system at risk of water hammer during a loss of offsite power event.

Description. Procedure STP-205-6301, Step 7.5, directed that the Inservice Test (IST) for the Division I emergency core cooling system keep fill Pump E21-PC002, be conducted before the IST of Check Valve E21-VF033. The test lineup for the pump resulted in pumping fluid at 27 gpm through the check valve for 15 to 20 minutes, which unintentionally flushed the valve immediately prior to its IST. Flushing the valve in this manner could remove corrosion products or other debris allowing the valve to pass its IST in the closed direction when it may not have passed otherwise.

Flushing Valve E21-VF033 prior to performing the IST was preconditioning because the potential problems with the valve could have been masked by this action. This was of particular concern because the valve was in a known degraded condition and had experienced two previous test failures due to corrosion buildup (see section 4OA2.e(2)(ii) of this report) . The team referenced NUREG 1482, “Guidelines for In-Service Testing at Nuclear Power Plants,” and NRC Information Notice 97-16, “Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing,” which provided guidance on what circumstances provided for acceptable versus unacceptable preconditioning. The team concluded that the preconditioning was not required for the protection of personnel or equipment, nor was it needed to meet manufacturer’s recommendations. Therefore, the preconditioning was unacceptable.

The valve’s safety function is to prevent backleakage during a design basis accident. During a loss of offsite power event, the keepfill pump loses power for up to 10 seconds. The valve must remain relatively leak tight to prevent a loss of water from the system discharge line. The loss of water would result in voiding near the system injection valve and, when the keepfill pump (or primary system pump) restarts, a water hammer could occur. Water hammer can challenge piping integrity and system operability.

Analysis. The failure to properly test the Check Valve E21-VF033 was a performance deficiency. The finding was more than minor because, if left uncorrected, the problem could result in a more significant safety concern. Specifically, the surveillance test may not identify valve failure. Using the Manual Chapter 0609, “Significance Determination Process,” Phase1 worksheet, the finding was of very low risk significance because it was not a design/qualification issue, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its technical specification allowable outage time, did not result in a loss of function of non safety-related risk significant equipment and was not risk significant due to external events. The finding had problem identification and resolution cross-cutting aspects because the licensee had failed to properly evaluate the issue as preconditioning in response to readily available industry information.

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," Appendix A, Section 8.b, recommends, in part, procedures for emergency core cooling system surveillance tests. Contrary to the above, since initial startup of the plant, procedure STP-205-6301 was inadequate because it provided steps that included preconditioning of Valve E21-VF033 prior to its surveillance. This issue is a violation of Technical Specification 5.4.1.a. Because this violation was of very low safety significance (Green) and was documented in the licensee's corrective action program as CR-RBS-2005-04123, it is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee planned to test Valve E21-VF033 in the as-found condition during the future surveillance activities. (NCV 50-458/2005008-01, Preconditioning of a Safety-related Valve Prior to Surveillance Testing)

ii. Failure to Replace Degraded Check Valve E21-VF033 in a Timely Manner

Introduction. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to take prompt corrective actions to replace low pressure core spray discharge line keep fill check Valve E21-VF033. The licensee had previously identified that Valve E21-VF033 had become degraded due to corrosion buildup.

Description. On October 29, 2003, and on September 28, 2004, Valve E21-VF033 failed to fully close during its IST. On both occasions, the licensee disassembled and inspected the valve, and identified that corrosion had accumulated on the valve internals causing the valve to not fully close. Following the September 28, 2004, valve failure, the licensee replaced the piston disc and spring and initiated Condition Report CR-RBS-2004-02799. As corrective measures, the licensee specified an additional valve internal inspection and, longer term, valve replacement. The additional inspection occurred on March 17, 2005. The inspection identified that the valve disc was not seating properly and that additional corrosion of the valve internals had occurred. The valve was again cleaned, reassembled and returned to service.

The acceptance criteria demonstrating the closure capability of Valve E21-VF033, as specified in Procedure STP-205-6301, was to verify that the low pressure core spray system discharge line low pressure alarm did not annunciate for at least 10 seconds after the keepfill pump was secured. The valve's safety function is to prevent back leakage during a design basis accident. During a loss of offsite power event, the keepfill pump loses power for up to 10 seconds. The valve must remain relatively leak tight to prevent a loss of water from the system discharge line. The loss of water would result in voiding near the system injection valve and, when the keepfill pump (or primary system pump) restarts, a water hammer could occur. Water hammer can challenge piping integrity and system operability. The team concluded that the repetitive corrosion buildup on the internal components of Valve E21-VF033 was a significant condition adverse to quality.

The team identified that the licensee failed to perform prompt corrective measures to address the significant condition adverse to quality. For example, the licensee's primary corrective action was to replace the valve. This corrective action was specified on October 13, 2004. Despite the fact that the replacement valve was in the warehouse and would take a maximum of 12 hours to install (licensee estimate), the licensee did not take advantage of at least three opportunities to implement the corrective measure. First, the licensee completed two low pressure core spray system "super-outages" - in January 2004 and again in August 2005. Finally, a refueling outage is scheduled for the Spring 2006. The licensee did not plant to replace the valve until after the outage in June 2006.

NRC Regulatory Information Summary, RIS 2005-020, "Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, Information to Licensees regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," provided the standard for dispositioning degraded and non-conforming conditions and stated that a degraded condition <u>should be resolved at the first available opportunity</u> unless an appropriate evaluation has been performed justifying a longer completion schedule. In this case, the licensee did not have adequate justification for the significant delay.

<u>Analysis</u>. The failure to take prompt corrective measures to address a significant condition adverse to quality was a performance deficiency. The finding was greater than minor because it was an equipment performance reliability issue which impacted the mitigating cornerstone objective to ensure the reliability of systems that respond to initiating events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was of very low risk significance because it was not a design/qualification issue, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its technical specification allowable outage time, did not result in a loss of function of non safety-related risk significant equipment and was not risk significant due to external events. This finding also had cross-cutting aspects in the area of problem identification and resolution because of the untimely corrective measures.

<u>Enforcement</u>. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that prompt corrective actions be taken to correct conditions adverse to quality. Contrary to the above, since September 28, 2004, when the licensee identified that Valve E21-VF033 had a ongoing problem with corrosion buildup, which caused valve failure (a significant condition adverse to quality), the licensee failed to take prompt corrective actions to address the problem. Because this violation was of very low safety significance (Green) and was documented in the licensee's corrective action program as CR-RBS-2005-04162, this violation is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. As an immediate corrective action, the licensee rescheduled the replacement of Valve E21-VF033 for February 2006. (NCV 50-458/2005008-02, Untimely Replacement of a Valve to Correct a Significant Condition Adverse to Quality).

iii. Failure to Maintain Minimum Critical Power Ratio Within Limits

Introduction. The team identified two examples of a Green non-cited Technical Specification 3.2.2, “Minimum Critical Power Ratio” (MCPR), violation for the failure to maintain the MCPR within the required limits during Operating Cycles 8 and 11. Fuel failures due to exceeding the limits were experienced during each cycle. Engineers failed to properly understand the affect of zinc injection on the cladding surfaces following the Cycle 8 fuel pin failures and zinc injection was reinitiated before the corrective actions to prevent recurrence were in place. The licensee had industry information that specified that zinc injection contributed to the accumulation of loose crud and the formation of tenacious crud on the fuel. The additional crud rendered the Technical Specifications MCPR calculations inaccurate.

Discussion

Cycle 8 Fuel Failures: During Operating Cycle 8, the licensee experienced seven fuel pin failures in high power regions of the core. All the fuel pin failures were located in first burn fuel bundles. During Refueling Outage 8, the licensee found a significant layer of crud on the fuel surface. Pictures of the crud indicated that it was primarily composed of loose iron oxide deposits but the team observed some tenacious crud on the cladding surface as well. The licensee did not perform a chemical analysis of the crud.

The crud increased the thermal resistance between the fuel cladding and the coolant such that cladding surface temperatures were substantially higher than would normally be expected. Normal cladding surface temperatures are about 560 EF (close to the bulk coolant temperature). General Electric (the fuel vendor) calculated that the cladding surface temperatures approached 1200 EF in localized areas. The higher temperatures increased the cladding oxidation rate and, at approximately 1 year into the cycle, the cladding oxidation layer extended the entire way through the cladding, creating a hole.

The team reviewed one technical study that discussed the behavior of crud on the surface of boiler tubes (“Two-Phase Flow and Heat Transfer,” D. Butterworth and G.F. Hewitt, Oxford University Press, 1977). The team noted that the thermal resistance of crud is not normally sufficient to cause cladding temperature increases consistent with those observed during Cycle 8. In most circumstances, “wick boiling” occurs within the crud. That is, capillary coolant channels within the crud deliver coolant to the cladding surface. Steam then escapes from the cladding surface in chimney type plumes. This is a fairly effective method of heat transfer. However, in some instances the capillary coolant channels can become clogged, creating a static steam blanket on the cladding surface. Steam is an exceptionally good thermal insulator. This is the process that caused the very high cladding surface temperatures and ultimately resulted in fuel cladding failure.

The team identified that the licensee had failed to maintain the Technical Specification 3.2.2, “Minimum Critical Power Ratio (MCPR)” within the required limits. The MCPR is the ratio of the fuel assembly power that would result in the onset of transition boiling to the actual fuel assembly power. A MCPR of 1.0 corresponds to the onset of transition boiling. Technical Specification 3.2.2 requires operators to maintain the MCPR within the limits specified by the Core Operating Limits Report (COLR). While the COLR allows MCPR to vary with reactor power (when greater than 23.8 % reactor power), it does not permit a MCPR of less than 1.08. MCPR limits are imposed to avoid fuel damage that can be caused by severe overheating of the cladding during both routine operations and anticipated operational occurrences (note: the MCPR contains margin, in part, to account for variations which may occur during anticipated operational occurrences). As specified in the Technical Specification 3.2.2 Bases, the operating MCPR limit is established to ensure that no fuel damage results during anticipated operational occurrences. Contrary to the above, as evidence by surface cladding temperatures that approached 1200 EF and the corresponding fuel damage, the licensee did not maintain MCPR within the required limits.

The MCPR limits were exceeded because the licensee did not account for the crud in their MCPR calculations and the crud affected the thermal resistance between the cladding surface and coolant. This additional thermal resistance rendered the licensee's MCPR calculations inaccurate and, consequently, the MCPR monitoring activities were ineffective.

Following the Cycle 8 cladding failures, the licensee did not definitely identify the root cause of the cladding failures. Instead, the licensee identified several chemistry events and recently implemented processes that could have caused, or contributed to, the cladding failures. Those potential causes included: 1) two conductivity excursions which occurred during Cycle 8 startups; 2) Cycle 8 was the first full cycle for zinc injection; 3) iron and copper levels were well above industry averages (River Bend has an Admiralty brass condenser, a known source of copper); and 4) the implementation of "maximum extended load line limit analysis, (MELLLA)" which permits operation at lower flow for a given power.

As initial corrective measures, the licensee: 1) managed plant operations to limit significant power transients, which can dislodge iron oxides in the plant and permit crud transfer to the fuel; 2) stopped zinc injection; 3) attempted to minimize the transport of copper to the reactor vessel through better chemistry controls; 4) temporarily discontinued MELLLA operations; and 5) planned to install a full flow filter, designed to filter out significant amounts of copper and iron from the condensate and feedwater stream. The licensee stated that the installation of the filter was the primary action to preclude repetition. While these interim corrective measures were in place, fuel failures did not repeat. The licensee installed the filter during Cycle 12.

Cycle 11 Fuel Failures: The team identified that the licensee had reversed one important corrective action (discontinuing zinc injection) BEFORE they installed the full flow filter (the corrective measure to prevent recurrence). For example, after about a 6 month delay in Cycle 10, the licensee reinitiated zinc injection. Since they had no fuel failures during that cycle, the licensee started zinc injection at the beginning of Cycle 11. In both instances the licensee initiated zinc injection without first fully understanding the potential impact that zinc had on the Cycle 8 fuel failures.

The licensee subsequently experienced additional fuel failures during Cycle 11. During Refueling Outage 11, the licensee inspected the fuel and found both fluffy and tenacious crud. The difference between Cycles 8 and 11 was that there was significantly more tenacious and less fluffy crud on the Cycle 11 fuel.

The licensee obtained a sample of the tenacious crud and performed an analysis. The analysis showed that the crud was composed primarily of iron oxide but contained significant amounts of zinc and copper. The analysis further explained that the zinc and copper formed oxides that clogged the capillary cooling channels within the crud. This permitted the formation of a static steam blanket, which elevated cladding temperatures significantly (just as in the Cycle 8 fuel failures). While the licensee did not estimate the actual cladding temperatures experienced during Cycle 11, the team determined that the behavior of the fuel failures in both cycles - in approximately the same high power locations and at about the same time (about a year into the cycles) - supports the assumption that maximum cladding temperatures were similar. As with the Cycle 8 fuel, the licensees MCPR calculations were rendered inaccurate by the crud.

Additional Information:

The team identified that, prior to zinc addition, the licensee knew that zinc addition had the potential to increase the formation of loose and tenacious crud on the fuel. Further, while the licensee established feedwater iron limits to avoid crud related problems, they did not adhere to the limits. Supporting information includes:

- The licensee had previously identified instances where other plants, who had already initiated zinc injection, had experienced crud buildup and the formation of tenacious crud on the fuel. The licensee's study "Impact of Zinc Injection on Fuel Performance," dated February 4, 1997, documented that Hope Creek and a foreign reactor (BWR 6) observed additional crud on fuel surfaces following the initiation of zinc injection. Further, Hatch and Perry had observed the formation of tenacious crud following zinc injection.

- The licensee did not maintain feedwater iron levels below previously identified thresholds that were established for zinc injection. Specifically, the licensee's study "Impact of Zinc Injection on Fuel Performance," stated, in part:

  > "The use of zinc injection, with a zinc level below 10 ppb in the reactor vessel and an [a] feedwater iron content less than 2 ppb, is also acceptable at River Bend."

  The 2 ppb feedwater iron limit was based on known industry experience. However, the licensee routinely operated with zinc injection while feedwater iron levels were well above the 2 ppb threshold. For example, the average Cycle 8 feedwater iron levels were 3.6 ppb while the average Cycle 11 feedwater iron levels were about 3 ppb.

In addition, the team noted that a number of studies on the River Bend Cycle 11 failures pointed to zinc addition as the cause for tenacious and loose crud buildup:

- Advanced Nuclear Technology Sweden AB Memo ANT 03-006M states, in part:

  > “The reason for RBS to have such high zinc injection rate is the high FW iron levels. The simultaneous high iron and zinc levels will, however, necessarily result in a very notable adherent and loose crud deposits...”

- “Fuel Failures During Cycle 11 at River Bend,” a paper written by Edward J. Ruzauskas of AREVA and David L. Smith of Entergy, states in part:

  > “Zinc addition in the coolant, while being beneficial to the general radiation levels during an outage, is noted to cause a tenacious crud deposition of zinc, iron, and nickel on the cladding.”

- “Performance of Framatome ANP BWR Fuel Rods,” a paper written by A. Seibold and R.S. Reynolds, Framatome ANP, states, in part:

  > “The high zinc levels, along with copper and silica, likely caused a formation of insulating strata in the CRUD structure, as evidenced by the CRUD flake analysis. The stratified CRUD lacks proper steam venting and resulted in localized steam blanketing of clad surface.”

Rod Bowing: The licensee identified one problem that was unique to the Cycle 11 failures - significant bowing of the failed fuel pins. This was caused by high temperatures over a larger area of the fuel pins. The cladding temperature had been sufficiently high to anneal the metal, change the micro structure of the zircaloy material. The minimum temperature for annealing zircaloy is about 930 EF. The team determined that the wide spread coverage of the tenacious crud likely caused this phenomena.

<u>Analysis.</u> The failure to maintain the MCPR within the required limits was a performance deficiency. The issue was more than minor because it was a primary chemistry control issue which affected the barrier integrity cornerstone objective to maintain the integrity of the fuel cladding. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 worksheet, the finding screened out as of very low safety significance (Green) because it only affected the fuel barrier. The issue had cross-cutting aspects in the areas of problem identification and resolution in that the licensee halted an interim corrective action (suspension of zinc injection) prior to the implementation of a final corrective action (installation of a feedwater water full flow filter) to prevent recurrence of the fuel cladding damage observed in cycle 8. The licensee re-initiated zinc injection during cycle 10 without fully understanding the ramifications of zinc injection on fuel cladding performance. This action resulted in fuel cladding damage during cycle 11.

<u>Enforcement.</u> Technical Specification 3.2.2 requires that, while greater than 23.8 percent reactor thermal power, operators maintain the MCPR greater than the operating limits specified in the COLR. Per the COLR, the lowest permissible MCPR is 1.08, but

this limit can be higher, depending on the power level. Contrary to the above, during Operating Cycles 8 and 11, the licensee operated the core outside of the specified MCPR limits, as evidence by excessively high cladding temperatures and fuel damage. Because this issue is of very low safety significance and has been entered into the licensee's corrective action program as CR-RBS-2006-0255, this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000458/2005008-03, Failure to Maintain MCPR within Operating Limits).

iv. Inadequate MOV Limit Switch Settings

Introduction. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion V (procedures) for the failure to set safety-related limit switches in accordance with documents appropriate to the circumstances for 34 safety-related MOV throttle valves. The licensee set MOV open indication light limit switches so that the open indication de-energized between the 95% and 100% closed positions, whereas the procedure and design drawing required that the limit switches be set to the 100% closed position. This practice had caused repetitive operational problems in the plant.

Discussion. The team performed a followup inspection to self-revealing Violation 50-458/2005004-03, which involved the failure to fully close a safety-related throttle MOV. On July 29, 2005, an operator had repositioned throttle Valve E12-MOVF037A to the closed position. When the valve indicated fully closed the operator released the hand switch, but the valve had not actually traveled all the way closed. On August 3, 2005, during operation of the residual heat removal system, water was inadvertently transferred from the suppression pool to the upper pool because Valve E12-MOVF037A was not fully closed. The licensee documented the problem in CR-RBS-2005-2772. The NRC report also noted that this was a repeat problem, as the same issue had occurred during a prior refueling outage. While following up to the more recent event, the licensee found an additional 8 throttle MOVs that appeared to be partially open.

As a corrective measure, the licensee advised operators to hold throttle valve hand-switches closed for an additional 5 seconds after receiving closed indication. The direction was provided to operators via a Night Order. In addition, the licensee updated a training plan to reflect the guidance. The licensee stated that, prior to the event, operators were trained to hold throttle valve switches in the closed position during normal operations training. However, based on the recurring events, the training was not effective.

The team identified that the licensee had taken inadequate corrective measures for the problem because they failed to identify and address the failure to set the limit switches in accordance with design documents and plant procedures. For example, the licensee routinely set the open limit switch lights to extinguish between 95% and 100% of the full closed position (when the light goes out it indicates that the valve is closed). However, the applicable Procedure GMP-0108, "VOTES Signature Testing of Gate, Glove and Torque Seated Butterfly Valves with Limitorque Actuators," Revision 4, Section 9, Acceptance Criteria stated, in part:

> "Verify limit switches are set per ESK [a design drawing] requirements AND the following:..
>
> 2. Open indicator should deenergize between 95% of full stroke and hardseat contact, C11."

The ESK design drawing required that the limit switches be set to the 100% closed position. The team noted that both of the above requirements could be met by setting the subject limit switches to the 100% closed position. In response to NRC questions, the MOV engineer stated that he did not check the ESK requirements. The throttle valves remained operable because operators were still required to hold the valve's closed for an additional 5 seconds following receipt of full closed indication. In total, 34 safety-related throttle valves were affected.

The team reviewed recommendations for the subject limit switch setting from the MOV actuator manufacturer. The recommendations indicated that the open limit switch be set to extinguish when the valve was fully closed. The team also contacted a technical representative from Limitorque. The team was advised that there is no technical reason why the licensee can't set the limit switches in accordance with the Limitorque recommendations.

<u>Analysis</u>. The failure to follow plant procedures and adjust MOV limit switches in accordance with design requirements was a performance deficiency. The issue was more than minor because it was a configuration control issue which affected the mitigating systems cornerstone objective to ensure the availability, reliability or function of a system or train in a mitigating system. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was of very low safety significance because it was a qualification deficiency which did not result in a loss of function and was not potentially risk significant due to external events. This finding had cross-cutting aspects in the areas of human performance for the failure to follow procedures. Additionally the issue had problem identification and resolution cross-cutting aspects because the licensee failed to identify the problem in response to a prior NRC violation.

<u>Enforcement</u>. 10 CFR 50, Appendix B, Criterion V (Procedures) states, in part "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." At River Bend, documents controlling MOV limit switch settings include Procedure GMP-0108, "VOTES Signature Testing of Gate, Globe and Torque Seated Butterfly Valves with Limitorque Actuators," Revision 4 and the applicable ESK (design drawing) for each valve. The documents specify that the limit switches for the safety-related open indication lights are required to be set at the full closed (100% closed) position. Contrary to the above, the licensee routinely set the limit switches so that the indication lights would go out when the throttle valves were in the 95% to 100% closed position. Because this issue is of very low safety significance and has been entered into the licensee's corrective action

program as CR-RBS-2005-4113, this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000458/2005008-04, Failure to Set MOV Limit Switches in Accordance with Design Documentation).

v. Adequacy of the Diesel Air Start System to Meet Design Requirements and Regulatory Guide 1.9 Commitments

Introduction. An Unresolved Item (URI) was opened pending the NRC's final evaluation of potential design discrepancies with the station's emergency diesel generators.

Discussion. During the inspection, the team identified the following discrepancies:

1. The Updated Safety Analysis Report indicated that the Division 1 and 2 emergency diesel generator air receivers were capable of providing 5 diesel generator starts. However, the licensee did not have testing or other analysis to demonstrate that the air receivers can provide 5 starts from the 160 psig minimum air receiver pressure permitted by the technical specifications. The licensee normally maintains the air receivers at approximately 250 psig.

   The inspectors noted that the licensee may have lost the five start capability when taking corrective measures for a prior problem. In 1991, the licensee changed the setpoint for the diesel air start "Cutout Interlock" for the diesel air start receivers from 150 psig to 120 psig to provide sufficient air pressure to allow one emergency start from 160 psig before the cutout interlock was actuated terminating the emergency start process. Justification for the change stated that there was no regulatory basis for the 150 psig value and that the cutout feature was not used by other manufacturers. The cutout interlock was incorporated in the diesel air start system due to the unique design of the licensee's start system. Unlike other diesel start systems common to the industry which secure the starting air after a set period of time, usually 3 to 10 seconds (if the diesel does not start), the design of the system at the River Bend Station ports starting air to the diesel until the diesel has started or the cutout interlock setpoint has been reached. According to manufacturer's data, the basis for selecting 150 psig as the cutout point was to preserve sufficient air capacity to ensure the capability for four additional starts. The data specified:

   > ... in the event the engine fails to start due to some malfunction, sufficient air remains in the single bank to start the engine four times, using NORMAL START at the local control panel. This is the purpose of the lockout feature; to conserve air in the event of an autostart failure."

   Technical Specification Bases 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," states in part:

...each diesel generator has an air start system with adequate capacity for five successive attempts on the diesel without recharging the air start receiver(s)."

Therefore, the Diesel Generators 1A and 1B starting air systems were sized to have the capacity for at least one emergency diesel generator start attempt above the air pressure interlock, and multiple manual start attempts below the interlock, without recharging its start receivers. For each diesel generator, either the forward or rear air start subsystem has the capacity to satisfy these multiple start requirements. The licensee believed that the diesel generators were only required to demonstrate one emergency start with multiple manual starts and that no specific number of multiple starts was required. Additional review by the team is required to determine the acceptability of the licensee's interpretation of the Updated Safety Analysis Report commitments. Specific information to review includes: 1) any test data or engineering calculations which demonstrate the diesel air start system is capable of performing 4 normal starts from 120 psig; and 2) information showing that the five start commitment has been removed from the licensing bases documents.

2. The inspectors identified that, in 2001, the licensee changed the status of the emergency diesel generator air compressors from safety-related to nonsafety-related without first performing a 50.59 evaluation. Once a 50.59 was performed, the inspectors further identified that the licensee did not consider the impact of the change on the over-speed trip function (a function described in Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used As Standby (Onsite) Electric Power Systems At Nuclear Power Plants." The air compressors are needed to maintain this function post-accident. The inspectors needed additional time to review the acceptability of this change without prior NRC approval. Specifically, the inspectors were concerned that by downgrading the compressors, the licensee was less likely to promptly address longstanding compressor performance problems (such as water in the compressor oil). Since the compressors are needed to support the over-speed trip function, a feature important to safety, it was not clear that the licensee would be able to conclude that the change would not result in more than a minimal increase in the probability of malfunction of equipment important to safety.

This is an unresolved item pending the completion of the additional NRC reviews (URI 05000458/2005008-05, Noted Design Discrepancies with the Diesel Generators).

Analysis. The potential issues associated with the design capabilities of the emergency diesel generators is under review by NRC staff. A determination of the safety significance of any performance deficiencies will be addressed in the resolution of the URI.

Enforcement. The potential issues associated with the design capabilities of the emergency diesel generators is under review by NRC staff. A determination of any enforcement aspects for any performance deficiencies will be addressed in the resolution of the URI.

4OA6 Exit Meeting

The team conducted an initial exit meeting on December 8, 2005, with Mr. R. King, Director, Nuclear Safety Assurance, and other members of the licensee's staff. After additional in-office inspection, the team conducted a final telephonic exit meeting on January 19, 2006, with Mr. King and other members of the licensee's staff. For each exit meeting the licensee acknowledged the findings. While some proprietary information was reviewed during the inspection, all proprietary information was returned to the licensee prior to the exit meeting.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Hinnenkamp	Vice President, Operations
B. Biggs	Coordinator, Safety and Regulatory Affairs
M. Davis	Supervisor, Radiation Control
C. Forpahl	Manager, Corrective Action and Assessment
H. Goodman	Director, Engineering
B. Houston	Manager, Maintenance
G. Huston	Assistant Operations Manager
K. Huffstatler	Technical Specialist, Licensing
D. Lorfing	Manager, Licensing
J. Maher	Superintendent, Reactor Engineering
P. Russell	Manager, System Engineering

NRC Personnel

P. Alter	Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000458/2005008-05	URI	Noted Design Discrepancies with the Diesel Generators (Section 4OA2.e(2)(v))

Opened and Closed

05000458/2005008-01	NCV	Unacceptable Preconditioning of a Safety-related Valve Prior to Surveillance Testing (Section 4OA2.e(2)(i))
05000458/2005008-02	NCV	Untimely Replacement of a Valve to Correct a Significant Condition Adverse to Quality (Section 4OA2.e(2)(ii))
05000458/2005008-03	NCV	Failure to Maintain MCPR within Operating Limits (Section 4OA2.e(2)(iii))
05000458/2005008-04	NCV	Failure to Set MOV Limit Switches in Accordance with Design Documentation (Section 4OA2.e(2)(iv))

Closed

None.

Discussed

None.

LIST OF ACRONYMS

ADAMS	Agency Document And Management System
EC	degrees Celsius
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
EF	Fahrenheit
FIN	finding
IST	Inservice Testing
LPCS	Low Pressure Core Spray
MCPR	Minimum Critical Power Ratio
MELLLA	maximum extended load line limit analysis
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
psig	pounds per square inch
PARS	Publicly Available Records System
URI	unresolved item

PARTIAL LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the team to accomplish the objectives and scope of the inspection and to support any findings:

Calculations

General Electric calculation of Cycle 8 fuel temperatures

G13.18.6.102-0; Analytical Trip Setpoint for RCIC/RHR Flow Transmitters E31-PDTN084A and B; January 20, 1987

12210-ES-211-0; Mass and Energy Release due to 4" RCIC Steam Line Break in RCIC Turbine Room with Friction Blowdown

Condition Reports (CR-RBS-)

2002-1621	2002-2184	2002-2525	2002-2622	2002-3695	2002-3691
2002-6447	2002-6448	2002-6449	2003-1584	2003-1827	2003-2133
2003-2677	2004-0029	2004-0318	2004-0531	2004-0475	2004-1567
2004-1734	2004-1751	2004-1775	2004-1779	2004-1780	2004-1785
2004-1863	2004-1915	2004-2078	2004-2328	2004-2398	2004-2400
2004-2469	2004-2996	2004-3100	2004-3023	2005-0647	2005-1418
2005-1602	2005-1735	2005-2138	2005-2722	2005-2772	2005-2836
2005-2887	2005-2888	2005-2889	2005-3126	2005-4128	2005-2169
2004-1389	2004-4218	2004-4219	2004-1855	2004-2906	2005-3127

2005-2202	2005-1748	2004-0762	2004-3160	2002-1243	2003-1118
2003-2054	2003-2082	2003-2437	2003-3203	2003-3344	2003-3456
2003-3457	2003-3486	2003-3607	2003-3740	2004-0126	2004-0389
2004-0671	2004-1061	2004-1717	2004-1724	2004-1813	2004-2128
2004-2144	2004-2290	2004-2316	2004-2332	2004-2799	2004-2828
2004-2841	2004-2913	2004-3100	2004-3203	2004-3566	2004-3709
2004-4203	2004-4289	2005-0140	2005-0145	2005-0167	2005-0822
2005-1043	2005-1045	2005-1102	2005-2292	2005-3503	2005-4162
2004-0183	2003-1213	2002-1714	2002-1779	2002-1769	2004-1839
2004-1858	2004-3518	2004-3546	2004-2759	2004-1761	2004-3518
2005-2836	2004-3551	2004-2759	2003-2955	2004-3077	2003-1540
2004-3895	2005-1475	2005-1480	2003-1213	2002-1380	2003-1016
2003-1016	2003-3110	2003-1205	2003-1178	2005-3151	2005-2727
2005-2843	2005-2836	2005-1405	2005-2173	2005-1684	2005-1680
2004-2165	2004-2604	2004-4403	2005-0276	2005-1011	2005-2490
2005-2760	2005-3131	2005-2624	2005-2836	2005-3156	2005-2843
2005-3165	2005-0024	2003-0911	2004-2141	2004-2757	2004-2165
2004-4279	2005-0276	2005-0366	2004-1793	2004-1797	2005-1619
2005-1101	2005-3131	2005-3471	2004-4064	2004-1209	2004-1213
2004-1483	2003-0911	2004-1711	2004-1986	2004-2306	2004-2630
2004-3715	2004-4207	2004-4467	2005-0276	2005-0757	2005-1163
2004-2165	2005-1604	2005-1946	2005-2176	2005-2943	2005-3131
2004-1703	2003-0911	2004-1951	2004-2273	2004-2141	2004-2603
2004-3715	2004-4207	2004-4467	2005-0276	2005-0757	2005-1163
2004-2165	2005-1551	2005-1946	2005-2176	2005-2525	2005-2943
2004-2165	2005-2598	2004-0169	2004-0191	2004-2544	2004-2165
2004-4404	2005-0558	2005-1011	2005-2490	2005-2760	2005-0378
2004-3162	2005-3131	2004-1964	2004-1839	2004-1858	2005-1450
2005-0516	2004-3895	2004-2165	2004-4197	2004-4203	2004-4225
2005-3114	2005-3154	2003-3132	2003-3133	2004-2160	2004-2165
2004-2756	2005-0365	2004-4249	2005-1965	2005-0820	2005-2943
2004-2077	2005-2727	2005-3256	2004-0704	2004-2707	2004-0704
2004-2220	2004-1287	2003-3367	2004-0668	2003-1213	1995-0339
1990-0558					

Root Causes (CR-RBS-)

1998-0720 2002-1911 2003-0218 2004-2332 2004-3546

Inspection Reports

05000458/2003003 05000458/2004003 05000458/2004005 05000458/2003005
05000458/2003009 05000458/2005002 05000458/2005003

Engineering Requests

Modification Request 86-0088; Revise Setpoint for RCIC High Steam Flow Isolation; February 25, 1986

Modification Request 86-1020; Add Time Delay Relays to Increase RCIC System Reliability; June 30, 1986

ER-RB-2004-0131; 480 V Safety-related GE AKR Circuit Breaker Replacement Modification

Engineering Request ER-RB-2004-0510-001

ER-RB-2000-0490-003 - Revised Feedwater Flow Uncertainty Analysis for the Ultrasonic Flow Meter

RBS-ER-98-0585, "Provide safety-related power for Div I and II forward start air compressors, after coolers, and dryers in emergency diesel generator air supply system."

RBS-ER-98-0426 50.59 "Evaluation for Upgrade of Division I and II diesel generator air start subsystems."

RBS-ER-99-003, "Failure modes and effects analysis for the standby diesel generator engine control system."

RBS-ER-2000-0090, "Installation of DC power available indicators on EGS-EG1A and B start air control circuit."

RBS-ER-98-0519-ERT-01, "Division I diesel generator low starting air pressure lockout control logic modification."

Drawings

ESK-06RHS11, "Elementary Diagram 480V Control Circuit Residual Heat Removal System," Revision 10

PID-27-06A; System 209 - Reactor Core Isolation Cooling; Revision 42

828E539AA; Elementary Diagram - Reactor Core Isolation Cooling; Revision 24

PID-27-05A, "Low Pressure Core Spray," Revision 23

Plant Procedures

ADM-0094, "MOV Periodic Verification Program," Revision 1

EIP-2-018, "Technical Support Center," Revision 28

EIP-2-020, "Emergency Operations Facility," Revision 27

EN-LI-102, "Corrective Action Process," Revision 2

GMP-0108, "VOTES Signature Testing of Gate, Glober and Torque Seated Butterfly Valves with Lmitorque Actuators," Revision 4

R-PL-026, “Respirator Protection Policy,” Revision 3

GMP-0099; Instrument Sensing Line High/Low Point Valves; Revision 4

SOP-0035; Reactor Core Isolation Cooling System (SYS #209); Revision 27

GEK-83376; Operating & Maintaining Instructions for Reactor Core Isolation Cooling Systems; January 4, 2005

STP-205-6301; LPCS Quarterly Pump and Valve Operability Test; Revision 14

ARP-601-21, Alarm No. 2318, "LPCS Injection Line Pressure Hi/Low," Revision 18

STP-205-4207, "LPCS Pump Discharge Pressure High/Low Channel Calibration Test, Revision 8A, performed on March 17, 2005

STP-205-6301, "LPCS Quarterly Pump and Valve Operability Test," Revision 14, performed on March 13, 2005

STP-205-6301, "LPCS Quarterly Pump and Valve Operability Test," Revision 14, performed on September 29, 2004

<u>Work Orders</u>

00045623-01 00061423-01 00068944-01 00069059-01 00053348-01 50983070-01

WO 00052423-01, "Repair E21-VF033 Leaks By Seat," performed September 29, 2004

<u>Other</u>

Operator Logs

Chemistry results for Cycles 8, 11, 12 and 13

Technical Specifications

Updated Safety Analysis Report

River Bend position paper on motor-operated valve limit switch settings

River Bend position paper on Cycle 8 and Cycle 11 fuel failures

Modification Request 96-0048, zinc injection skid

Advanced Nuclear Technology Sweden AB Memo ANT 03-0006M

“Fuel Failures During Cycle 11 at River Bend,” September 19, 2004

“Impact of Zinc Injection on Fuel Performance,” February 4, 1997

“TMI-1 Cycle 10 Fuel Rod Failures,” Electric Power Research Institute (EPRI), October, 1998

“Crud-induced Cladding Corrosion Failures in TMI-1 Cycle 10,” R. Tropasso (Exelon Corp), J. Willse (Framatome ANP), & B. Cheng (EPRI), September 19, 2004

“Performance of Framatome ANP BWR Fuel Rods,” A. Seibold (Framatome ANP), and R.S. Reynolds (Framatome ANP), September 19, 2004

“An Integrated Approach to Maximizing Fuel Reliability,” EPRI, September 19, 2004

“Two Phase Flow and heat Transfer,” D. Butterworth and G. F. Hewitt, Oxford Press, 1977

“BWR Feul Deposit Sample Evaluation, River Bend Cycle 11 Crud Flakes,” EPRI

“Summary of Meeting Between the Nuclear Regulatory Comission (NR) Staff and Entergy Operations, INC. (EOI), River Bend Station Management,” June 22, 1999

“Grand Gulf Station Engineering Standard for Motor-Operated Valve Wiring and Lmit Switch Control,” Revision 2

RLP-LOR-1007, License Operator Requalification Training, Revision 1

Limitorque Type SMB Instruction and Maintenance Manual

McGraw-Edison RHF-90 with Type OA-4 hydraulic operating mechanism vendor manual

EPRI NP-6766, “Water Hammer Prevention, Mitigation, and Accommodation,” July 1992

System Performance Indicators

List of respirator qualified health physics technicians and training records

NRC Regulatory Guide 1.27; Ultimate Heat Sink for Nuclear Plants - Draft; January 1976

SDC-209; Reactor Core Isolation Cooling System Design Criteria System Number 209; Revision 3

Repetitive Task CKTBRK; Inspect AKR 30/50 with ECS Device

General Electric Services Information Letter 448; Maintenance and Lubricants for GE Type AK/AKR Circuit Breakers; Revision 2

NUREG-1482; Guidelines for Inservice Testing at Nuclear Power Plants; Revision 1

Cycle 11 MFLCPR History Report

EN-HU-101, "Human Performance Program," Revision 0

EN-LI-118, "Root Cause Analysis Process," Revision 1

LAR-2001-026 UHS Makeup Water

LER 2003-006-00

LER 2003-008-00

LER 2004-001-00

LER 2004-002-00

LER 2004-005-01

LER 2005-001-01

LER 2005-002-00

LER 2004-003-01

LER 2004-004-01

Maintenance Rule Function RBS-1-F-205-06, "Maintain pump discharge line full of water to minimize injection time and to prevent water hammer," Dated November 14, 2005

Setpoint Data Sheet Number 12210-PN-E21-PISN654, "Alarm On Low Pressure In Discharge Line Fill System," Dated May 12, 1993

TRM 5.5.6, "Inservice Inspection and Testing Programs"

USAR 6.3.2.2.5, "ECCS Discharge Line Fill System"

Technical Specification 3.7.1, "Standby Service Water and Ultimate Heat Sink

RBS Updated Safety Analysis Report rev 12 dated December 1999

RBS Safety Evaluation Report dated May 1984

Regulatory Guide 1.9, Rev. 2, “Selection, design, and qualification of diesel-generator units used as standby (onsite) electric power systems at nuclear power plants.”

Regulatory Guide 1.75 rev 3, “Criteria for independence of electrical safety systems.”

Various Security logs 10 Quarters

10CFR21-0090 rev 0 dated 09/16/05, “Governor drive coupling element P/N AK-007-001.”

RF-11 Reactor Reassembly Radiological Work Plan

Information Request 1 - August 2005
River Bend Station PIR Inspection (IP 71152; Inspection Report 50-458/05-08)

The inspection will cover the period of September 1, 2003 to September 30, 2005. All requested information should be limited to this period unless otherwise specified. To the extent possible, please provide the information in electronic media in the form of CDs. The agency's document software is Corel Wordperfect 10, Presentations, and Quattro Pro. However, we can also accept Microsoft suite files and Adobe Acrobat (.pdf) text files.

Please provide the following information to the following address by September 9, 2005.

NRC Resident Inspector Office
ATTN: Zachary Dunham
P.O. Box 69
Richland, WA. 99354

Note: On summary lists, please include a description of the problem, status, and initiating date.

1. Summary list of all condition reports related to significant conditions adverse to quality that were opened or closed during the period

2. Summary list of all condition reports related to conditions adverse to quality that were opened or closed during the period

3. Summary lists of all condition reports which were up-graded or down-graded during the period

4. A list of all corrective action documents that subsume or "roll up" one or more smaller issues for the period

5. Summary lists of operator workarounds, engineering review requests and/or operability evaluations, temporary modifications, and control room and safety system deficiencies opened or closed during the period.

6. List of all root cause analyses completed during the period

7. List of root cause analyses planned, but not complete at the end of the period

8. List of plant safety issues raised or addressed by the employee concerns program

9. List of action items generated or addressed by the plant safety review committees during the period

10. All quality assurance audits and surveillances of corrective actions completed during the period

11. All corrective action activity reports, functional area self-assessments, and non-NRC third party assessments completed during the period (do not include INPO assessments)

12. Corrective action performance trending/tracking information generated during the period and broken down by functional organization

13. Governing procedures/policies/guidelines for:

    a. Corrective action program/condition reports
    b. Apparent and root cause evaluation/determinations
    c. Employee concerns program

14. A listing of all external events evaluated for applicability at River Bend Station during the period

15. Condition reports or other actions generated during the period for each of the items below:

    a. Part 21 reports
    b. NRC Information Notices, Bulletins, and Generic Letters
    c. LERs issued by River Bend Station
    d. GE SILs
    e. NCVs and Violations issued to River Bend Station

16. Security event logs and security incidents during the period

17. Radiation protection event logs during the period

18. Condition reports generated as a result of emergency planning drills and tabletop exercises during the period

19. Current system health reports or similar information during the period

20. Condition reports associated with maintenance preventable functional failures during the period

21. Condition reports associated with adverse trends in equipment, processes, procedures, or programs during the period

22. Corrective action effectiveness review reports generated during the period

23. List of emergency plan exercise and drill deficiencies during the period

24. List of training deficiencies, requests for training improvements, and simulator deficiencies for the period

Information Request 2 - October 2005
River Bend Station PIR Inspection (IP 71152; Inspection Report 50-458/05-08)

The inspection will cover the period of September 1, 2003 to September 30, 2005. All requested information should be limited to this period unless otherwise specified. To the extent possible, please provide the information in electronic media in the form of CDs. The agency's document software is Corel Wordperfect 10, Presentations, and Quattro Pro. However, we can also accept Microsoft suite files and Adobe Acrobat (.pdf) text files.

1. Detailed evaluations of the GE SILs provided in information request 1.

2. Quality assurance audit reports during the period.

3. Copies of corrective action documents associated with the safety committee action items provided in information request 1.

4. Copies of the following documents including detailed documentation of the issue, resolution, corrective actions, and final disposition as applicable.

Corrective Action Documents

CR-RBS-2004-03162	CR-RBS-2005-00378	CR-RBS-2002-02088
CR-RBS-2003-01016	CR-RBS-2003-01178	CR-RBS-2003-01205
CR-RBS-2003-01016	CR-RBS-2002-01380	CR-RBS-2003-01213
CR-RBS-2003-01213	CR-RBS-2003-02368	CR-RBS-2003-01894
CR-RBS-2003-02054	CR-RBS-2003-02437	CR-RBS-2003-01213
CR-RBS-2003-03042	CR-RBS-2002-01372	CR-RBS-2003-03515
CR-RBS-2003-02955	CR-RBS-2003-01540	CR-RBS-2002-01714
CR-RBS-2002-01769	CR-RBS-2002-01779	CR-RBS-2004-01083
CR-RBS-2003-02082	CR-RBS-2004-00924	CR-RBS-2004-01076
CR-RBS-2004-01856	CR-RBS-2004-01839	CR-RBS-2004-01858
CR-RBS-2004-01567	CR-RBS-2004-01893	CR-RBS-2004-03395
CR-RBS-2004-04296	CR-RBS-2004-03456	CR-RBS-2004-03551
CR-RBS-2004-03518	CR-RBS-2004-04218	CR-RBS-2004-03203
CR-RBS-2004-01287	CR-RBS-2004-02759	CR-RBS-2004-01761
CR-RBS-2004-00276	CR-RBS-2004-00771	CR-RBS-2004-02334
CR-RBS-2005-01726	CR-RBS-2005-02269	CR-RBS-2005-01400
CR-RBS-2005-01475	CR-RBS-2005-01480	CR-RBS-2003-03203
CR-RBS-2004-02332	CR-RBS-2004-02841	CR-RBS-2004-03518
CR-RBS-2004-03546	CR-RBS-2004-04289	CR-RBS-2005-00140
CR-RBS-2005-02292	CR-RBS-2003-03367	CR-RBS-2003-03424
CR-RBS-2004-00668	CR-RBS-2004-01317	CR-RBS-2004-02759
CR-RBS-2004-02842	CR-RBS-2004-02906	CR-RBS-2004-03077
CR-RBS-2004-03566	CR-RBS-2004-03895	CR-RBS-2004-04203
CR-RBS-2004-04218	CR-RBS-2005-00822	CR-RBS-2005-01141
CR-RBS-2005-01481	CR-RBS-2003-03110	CR-RBS-2003-03787
CR-RBS-2004-00455	CR-RBS-2004-01109	CR-RBS-2004-01270
CR-RBS-2004-01497	CR-RBS-2004-02486	CR-RBS-2004-02537

CR-RBS-2004-02772	CR-RBS-2004-02799	CR-RBS-2004-02869
CR-RBS-2004-03029	CR-RBS-2004-03129	CR-RBS-2004-03160
CR-RBS-2004-03170	CR-RBS-2004-03435	CR-RBS-2004-03709
CR-RBS-2004-04064	CR-RBS-2004-01923	CR-RBS-2004-02165
CR-RBS-2005-02598	CR-RBS-2005-02608	CR-RBS-2005-02641
CR-RBS-2005-02643	CR-RBS-2005-02836	CR-RBS-2005-02883
CR-RBS-2005-02959	CR-RBS-2005-03111	CR-RBS-2005-02680
CR-RBS-2005-03114	CR-RBS-2005-03154	CR-RBS-2005-03122
CR-RBS-2005-03214	CR-RBS-2005-03127	CR-RBS-2005-03168
CR-RBS-2005-03329	CR-RBS-2005-02447	CR-RBS-2005-02624
CR-RBS-2005-03248	CR-RBS-2005-03308	CR-RBS-2005-03279
CR-RBS-2005-02975	CR-RBS-2005-03296	CR-RBS-2005-02727
CR-RBS-2004-01567	CR-RBS-2004-01893	CR-RBS-2004-02332
CR-RBS-2004-01845	CR-RBS-2004-02165	CR-RBS-2004-03545
CR-RBS-2004-03895	CR-RBS-2005-00564	CR-RBS-2004-02192
CR-RBS-2004-02193	CR-RBS-2005-00760	CR-RBS-2005-00167
CR-RBS-2004-02252	CR-RBS-2003-00911	CR-RBS-2004-02141
CR-RBS-2003-00534	CR-RBS-2004-01158	CR-RBS-2005-01602
CR-RBS-2005-03126	CR-RBS-2005-02252	CR-RBS-2005-02646
CR-RBS-2005-02789	CR-RBS-2004-03811	CR-RBS-2005-01309
CR-RBS-2005-00419	CR-RBS-2005-00024	CR-RBS-2004-01615
CR-RBS-2005-00124	CR-RBS-2005-00940	CR-RBS-2005-02843
CR-RBS-2003-03038	CR-RBS-2003-03042	CR-RBS-2003-03132
CR-RBS-2003-03133	CR-RBS-2003-03770	CR-RBS-2003-03764
CR-RBS-2004-00169	CR-RBS-2004-00191	CR-RBS-2004-00183
CR-RBS-2004-00479	CR-RBS-2004-00346	CR-RBS-2004-01209
CR-RBS-2004-01213	CR-RBS-2004-01483	CR-RBS-2003-00911
CR-RBS-2004-01711	CR-RBS-2004-01986	CR-RBS-2004-02306
CR-RBS-2004-02630	CR-RBS-2004-03715	CR-RBS-2004-04207
CR-RBS-2004-04467	CR-RBS-2005-00276	CR-RBS-2005-00757
CR-RBS-2005-01163	CR-RBS-2004-02165	CR-RBS-2005-01604
CR-RBS-2005-01946	CR-RBS-2005-02176	CR-RBS-2005-02943
CR-RBS-2005-03131	CR-RBS-2004-01239	CR-RBS-2003-02743
CR-RBS-2005-03511	CR-RBS-2005-03183	CR-RBS-2004-01345
CR-RBS-2004-00854	CR-RBS-2004-01394	CR-RBS-2004-01083
CR-RBS-2002-00847	CR-RBS-2004-02992	CR-RBS-2004-04218
CR-RBS-2005-00329	CR-RBS-2005-00285	CR-RBS-2005-00338
CR-RBS-2005-00348	CR-RBS-2005-00467	CR-RBS-2005-00495
CR-RBS-2005-03350	CR-RBS-2004-01557	CR-RBS-2004-01560
CR-RBS-2004-01558	CR-RBS-2004-01559	CR-RBS-2004-01703
CR-RBS-2003-00911	CR-RBS-2004-01951	CR-RBS-2004-02273
CR-RBS-2004-02141	CR-RBS-2004-02603	CR-RBS-2004-03715
CR-RBS-2004-04207	CR-RBS-2004-04467	CR-RBS-2005-00276
CR-RBS-2005-00757	CR-RBS-2005-01163	CR-RBS-2004-02165
CR-RBS-2005-01551	CR-RBS-2005-01946	CR-RBS-2005-02176
CR-RBS-2005-02525	CR-RBS-2005-02943	CR-RBS-2005-03131
CR-RBS-2005-03471	CR-RBS-2004-01793	CR-RBS-2004-01797
CR-RBS-2004-01893	CR-RBS-2004-01567	CR-RBS-2004-01964
CR-RBS-2004-01839	CR-RBS-2004-01858	CR-RBS-2004-02077

CR-RBS-2004-00704	CR-RBS-2004-02707	CR-RBS-2004-02160
CR-RBS-2004-02165	CR-RBS-2004-02756	CR-RBS-2004-04249
CR-RBS-2005-00365	CR-RBS-2005-00820	CR-RBS-2005-01965
CR-RBS-2005-02943	CR-RBS-2005-03256	CR-RBS-2004-02193
CR-RBS-2004-02192	CR-RBS-2004-02220	CR-RBS-2004-00704
CR-RBS-2004-02453	CR-RBS-2004-02454	CR-RBS-2004-02478
CR-RBS-2004-02544	CR-RBS-2004-02165	CR-RBS-2004-04404
CR-RBS-2005-00558	CR-RBS-2005-01011	CR-RBS-2005-02490
CR-RBS-2005-02760	CR-RBS-2005-03131	CR-RBS-2004-02604
CR-RBS-2004-02165	CR-RBS-2004-04403	CR-RBS-2005-00276
CR-RBS-2005-01011	CR-RBS-2005-02490	CR-RBS-2005-02760
CR-RBS-2005-03131	CR-RBS-2004-02624	CR-RBS-2004-02408
CR-RBS-2004-02654	CR-RBS-2005-03440	CR-RBS-2005-03463
CR-RBS-2005-03482	CR-RBS-2004-02757	CR-RBS-2004-02165
CR-RBS-2004-04279	CR-RBS-2005-00276	CR-RBS-2005-00366
CR-RBS-2004-02867	CR-RBS-2004-02842	CR-RBS-2004-02877
CR-RBS-2004-02842	CR-RBS-2004-03289	CR-RBS-2004-03272
CR-RBS-2004-03520	CR-RBS-2004-03518	CR-RBS-2004-03565
CR-RBS-2004-03566	CR-RBS-2004-03592	CR-RBS-2004-03760
CR-RBS-2004-03709	CR-RBS-2004-04018	CR-RBS-2004-04041
CR-RBS-2004-04051	CR-RBS-2004-04050	CR-RBS-2004-04062
CR-RBS-2004-04197	CR-RBS-2004-04203	CR-RBS-2004-04225
CR-RBS-2004-04218	CR-RBS-2004-04219	CR-RBS-2005-00124
CR-RBS-2005-00112	CR-RBS-2004-04238	CR-RBS-2004-04218
CR-RBS-2004-04252	CR-RBS-2004-04203	CR-RBS-2004-04426
CR-RBS-2004-02069	CR-RBS-2005-00518	CR-RBS-2005-00519
CR-RBS-2005-00124	CR-RBS-2004-01083	CR-RBS-2005-00364
CR-RBS-2005-01045	CR-RBS-2004-02799	CR-RBS-2005-01450
CR-RBS-2005-00516	CR-RBS-2005-01619	CR-RBS-2005-01101
CR-RBS-2005-01684	CR-RBS-2005-01680	CR-RBS-2005-01405
CR-RBS-2005-02173	CR-RBS-2005-02124	CR-RBS-2005-02178
CR-RBS-2005-02329	CR-RBS-2005-02275	CR-RBS-2005-02480
CR-RBS-2005-02255	CR-RBS-2005-02652	CR-RBS-2005-02918
CR-RBS-2005-02720	CR-RBS-2005-02721	CR-RBS-2005-02773
CR-RBS-2005-02843	CR-RBS-2005-02836	CR-RBS-2005-03076
CR-RBS-2005-02946	CR-RBS-2005-03126	CR-RBS-2005-01602
CR-RBS-2005-03151	CR-RBS-2005-02727	CR-RBS-2005-03156
CR-RBS-2005-02836	CR-RBS-2005-03165	CR-RBS-2004-03525
LO-OPX-2005-00122	LO-OPX-2005-00093	LO-OPX-2005-02169
LO-OPX-2003-00241	LO-OPX-2003-00370	LO-OPX-2003-00401
LO-OPX-2004-00002		

Employee Concerns Program Files

No. 04-03-01	No. 04-06-01

Third Information Request

Questions provided to the licensee in writing following the onsite inspection:

- How many fuel bundles are in the River Bend core?
- What was the total crud loading (in pounds) in Cycles 8 and 11? There was a statement that there was an additional 700 pounds of crud in the reactor vessel than was expected.
- What chemical was involved in the chemical transient that occurred at the start of Cycle 8?
- Please provide the operator logs from the change to Mode 2 in cycle 8 to 4 days after achieving 100% power.
- Please provide a copy of NEDO-32456.
- One of your analysis for zinc injection specified maintaining less than 2 ppb feedwater iron. Did anyone do any additional analysis to account operation with feedwater iron levels that were above this limit?
- How were the affects from feedwater copper factored into the zinc recommendations?
- For the MOV issue, did the site modify the simulator so that (if an operator forgot to hold the control switch down an extra 5 seconds for a throttle valve) the simulator would simulate some flow through the valve?
- When did Doug (the MOV supervisor) go through operations training? He had stated that, when he went through operations training, he was trained to hold the valve closed for an extra five seconds.
- Please provide the "limit switch development chart" for Valve E12MOVF037A.