November 8, 2004

EA-03-131

Mr. Mark B. Bezilla
Vice President-Nuclear, Davis-Besse
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2004014

Dear Mr. Bezilla:

On September 30, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Davis-Besse Nuclear Power Station. The enclosed inspection report documents the inspection findings which were discussed on September 29, 2004, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

For the entire inspection period, the Davis-Besse Nuclear Power Station was under the Inspection Manual Chapter (IMC) 0350 Process. The Davis-Besse Oversight Panel assessed inspection findings and other performance data to determine the required level and focus of followup inspection activities and any other appropriate regulatory actions. Even though the Reactor Oversight Process had been suspended at the Davis-Besse Nuclear Power Station, it was used as guidance for inspection activities and to assess findings.

Based on the results of this inspection, the NRC has determined that violations of NRC requirements occurred. The report documents one inspector-identified finding and three self-revealed findings of very low safety significance (Green), all of which involved violations of NRC requirements. The findings did not present any immediate safety concerns. Because the violations were of very low safety significance and because the issues were entered into your corrective action program, the NRC is treating each of the findings as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

Section 4OA2 of the attached report documents an NCV for inadequate corrective actions associated with repetitive operational performance issues. Inspection Report 05000346/2003011 documented a similar issue. The current report also documents that in some cases, corrective actions and effectiveness reviews were extended significantly beyond original completion schedules. Due to the repeat nature of the operational performance issues, we request that you provide, within 30 days of the date of this letter, a written response detailing the corrective actions that have been implemented to avoid further similar violations. Your response should include the basis for your belief that these corrective actions will be effective at preventing future operations department performance errors and your assessment of the impact that deferral of corrective actions and effectiveness reviews has had on performance improvement in operations. Your response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; and the NRC Resident Inspector at Davis-Besse.

If you contest the severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-001; and the NRC Resident Inspector at Davis-Besse. at Davis-Besse.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) 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 by Christine A. Lipa for/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

Docket No. 50-346
License No. NPF-3

Enclosure: Inspection Report 05000346/2004014
w/Attachment: Supplemental Information

See Attached Distribution

DOCUMENT NAME: E:\Filenet\ML043270433.wpd

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII		RIII					
NAME	Clipa:dtp		JGrobe					
DATE	11/08/04		11/08/04					

OFFICIAL RECORD COPY

cc w/encl: The Honorable Dennis Kucinich
G. Leidich, President - FENOC
J. Hagan, Senior Vice President
Engineering and Services, FENOC
L. Myers, Chief Operating Officer, FENOC
Plant Manager
Manager - Regulatory Compliance
M. O'Reilly, Attorney, FirstEnergy
Ohio State Liaison Officer
R. Owen, Administrator, Ohio Department of Health
Public Utilities Commission of Ohio
President, Board of County Commissioners
of Lucas County
C. Koebel, President, Ottawa County Board of Commissioners
D. Lochbaum, Union Of Concerned Scientists
J. Riccio, Greenpeace
P. Gunter, N.I.R.S.

ADAMS Distribution:
AJM
DFT
SPS1
RidsNrrDipmIipb
GEG
HBC
CST1
CAA1
C. Pederson, DRS (hard copy - IR's only)
DRPIII
DRSIII
PLB1
JRK1
DB0350
ROPreports@nrc.gov (inspection reports, final SDP letters, any letter with an IR number)

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2004014
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	August 15 through September 30, 2004
Inspectors:	S. Thomas, Senior Resident Inspector J. Rutkowski, Resident Inspector M. Salter-Williams, Resident Inspector D. Passehl, Project Engineer J. Jacobson, Senior Reactor Inspector G. Wright, Project Engineer
Observers:	R. Smith, Reactor Engineer A. Garmoe, Reactor Engineer
Approved by:	C. Lipa, Chief Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000346/2004014; 8/15/2004 - 9/30/2004; Davis-Besse Nuclear Power Station; Maintenance Risk Assessment and Emergent Work Evaluation, Problem Identification and Resolution, Event Followup.

This report covers a 7 week period of resident inspection. The inspection was conducted by Region III inspectors and resident inspectors. Four Green findings associated with four non-cited violations were identified. 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.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was self-revealed when control room staff attempted to add water to the makeup tank using equipment that had been removed from service as part of the clearance which supported work on makeup system valves MU362 and MU363. The control room staff was unaware of the status of the normal makeup water sources to the reactor coolant system, even though the system's status was clearly documented in the Limiting Condition for Operation Tracking Log, a document which is required to be reviewed by the Shift Manager, the Unit Supervisor, and the Reactor Operator prior to shift turnover.

  The inspectors concluded that the finding was more than minor because the operator's lack of knowledge of system status challenged their ability to adjust control rod index by adding water to the reactor coolant system and to perform selected abnormal operating procedures prepared to address small reactor coolant system leaks. This finding was of very low safety significance because, during the time period the clearance impacted the operation of the makeup water sources, neither the ability to control makeup tank water level or to maintain an appropriate rod control index were challenged. This was determined to be a Non-Cited Violation of Technical Specification 6.8.1.a. (Section 1R13)

Cornerstone: Mitigating System

- Green. A finding of very low safety significance was self-revealed when the licensee discovered, during planned work activities, that the Steam Feedwater Rupture Controls System logic cards could energize in a blocked condition after being de-energized. This condition could prevent automatic isolation of a faulted number 2 steam generator concurrent with a loss of offsite power. This condition was introduced into the system logic subsequent to a design change completed on the Steam Feedwater Rupture Controls System in 1988. When recognized in 2003, the licensee corrected the design deficiency.

The inspectors concluded that the finding was greater than minor because it involved the attributes of design control and equipment reliability and could have affected the mitigating systems objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because it did not result in an actual loss of safety function since the starting of the auxiliary feedwater system was not affected and a faulted number 2 steam generator could be isolated with operator action if automatic isolation did not occur. This was determined to be a Non-Cited Violation of Technical Specification 3.3.2.2. (Section 4OA3)

- Green. A finding of very low safety significance was self-revealed during the licensee's inspection activities to address emergency core cooling system deficiencies, documented in LER 50-346/2002-005, and involved the licensee's failure to effectively implement corrective actions to verify the adequacy of the design of the containment emergency sump screen, and to implement effective corrective actions to repair an existing gap. Based on the inspectors' analysis, it was unlikely that the ECCS would be impacted.

  The inspectors concluded that the finding was greater than minor because the gap was associated with the objective and attributes of the Mitigating Systems and the Barrier Cornerstones. Specifically, the containment emergency sump screen was sized to pass no more than 1/4-inch debris particles and debris larger than 1/4-inch could have potentially damaged emergency core cooling system (ECCS) equipment and/or clogged the containment spray system (CSS) nozzles. The finding was of very low safety significance because: 1) the gap was not a design or qualification deficiency which resulted in a loss of function per Generic Letter 91-18, Revision 1; 2) did not represent an actual loss of safety function of a mitigating system; 3) did not represent an actual loss of safety function of a single train of a mitigating system for greater than its Technical Specification allowed outage time; 4) did not represent an actual loss of safety function of one or more non-Technical Specification mitigating system trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours; and 5) did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event. In addition, containment spray is not a large early release frequency contributor per IMC 0609 Appendix H. This was determined to be a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions. (Section 4OA3)

Cornerstone: NA:

- Green. The inspectors identified a finding having very low safety significance regarding the licensee's failure to identify proper corrective actions to preclude repetition of conditions adverse to quality as required by the Corrective Action Program. Specifically, corrective actions implemented to address repetitive Technical Specification violations and licensed operator performance errors, were not effective in precluding recurrence of similar events.

  The finding was more than minor because, if left uncorrected, the issue would become a more significant safety concern. Because this finding did not directly affect any of the cornerstone attributes, it was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c. The finding was determined to be of very low safety significance because no safety systems were degraded nor was any safety equipment rendered inoperable directly due to this issue. The issue was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings” which requires that activities affecting quality shall be accomplished in accordance with prescribed instructions and procedures. (Section 4OA2)

B. Licensee Identified Findings

None

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, the plant was operating at approximately 100 percent power. During this inspection period, brief planned power reductions of less than 10 percent occurred on two occasions to support planned testing. On each occasion, the testing was completed and power was restored to approximately 100 percent. The plant operated at approximately 100 percent power for the remainder of the inspection period.

For the entire inspection period, the Davis-Besse Nuclear Power Station was under the IMC 0350 Process.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment

.1 Partial Walkdowns (71111.04Q)

a. Inspection Scope

The inspectors verified equipment alignment to identify any discrepancies that would impact the function of system components. The inspectors also verified that the licensee had properly identified and resolved any equipment alignment problems that would cause initiating events or impact the availability and functional capability of the mitigating system. Documentation reviewed as part of this inspection included plant procedures, drawings, and the Updated Safety Analysis Report (USAR), to determine the correct system lineup.

During the walkdown, the inspectors also evaluated the material condition of the equipment to verify that there were no significant conditions not already in the licensee's corrective action system. The following two samples were selected:

- decay heat injection pump 1 during a planned train 2 work outage; and
- station black out diesel generator.

b. Findings

No findings of significance were identified.

.2 Complete Walkdowns (71111.04S)

The inspectors verified equipment alignment to identify any discrepancies that impacted the function of system components within the Auxiliary Feedwater System. The inspectors also verified that the licensee had properly identified and resolved any equipment alignment problems that would cause initiating events or impact the availability and functional capability of the mitigating system. Documentation reviewed as part of this inspection included plant procedures, drawings, and the Updated Safety Analysis Report (USAR), to determine the correct system lineup. Additionally, the inspectors evaluated outstanding maintenance work requests and condition reports to identify any deficiencies that could affect the ability of the system to perform its design basis function. A majority of the inspectors' time was spent performing a walkdown inspection of the system. Key aspects of the walkdown inspection included verifying:

- valves were correctly positioned and did not exhibit leakage that would impact their functionality;
- electrical power was available as required;
- major system components were correctly labeled, lubricated, cooled, and ventilated;
- hangers and supports were correctly installed and functional;
- essential support systems were operational;
- ancillary equipment or debris did not interfere with system performance; and
- valves were locked as required by the licensee's locked valve program.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection inspections focused on the availability, accessibility, and condition of fire fighting equipment, the control of transient combustibles, and the condition and status of installed fire barriers. The inspectors selected fire areas for inspection based on their overall contribution to internal fire risk, as documented in the Individual Plant Examination of External Events, and their potential to impact equipment which could initiate a plant transient. Inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use, that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits, and that fire doors, dampers, and penetration seals appeared to be in satisfactory condition.

The following one area was inspected:

- Fire Area T (component cooling water heat exchanger and pump room)

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Integrated Control System

a. Inspection Scope

The inspectors evaluated the licensee's handling of performance issues associated with the integrated control system (ICS) including the system interfaces with turbine generator, feedwater system, and control rod drive system. This inspection consisted of evaluating the following licensee activities:

- work scheduling practices, including consideration of risk of transient initiation while performing work on operating components;
- use of the condition report process and work order notification system in identifying deficiencies and issues with the equipment;
- problem solving and issue resolution associated with the failures and degradations of components associated with the ICS;
- that maintenance activities on the components had been assigned appropriate risk classification;
- that goals and corrective actions for the long term reliability were appropriate;
- that short term corrective actions were appropriate for deficiencies with potential to become operator workarounds or the potential to become transient initiators; and
- that maintenance rule system status determination was appropriate for the equipments' recent history and current open work items.

With regard to long-term reliability, the inspectors also reviewed the potential effect of deferring until June 30, 2005, an investigation recommended in October 2002, for improvements in either the reliability or replaceability of auxiliary and transfer relays in ICS. Several ICS relay modules were replaced because of relay problems since restart of the unit in 2004. Concurrently, the inspectors reviewed the affect of deferring clarification of ICS module preventive maintenance requirements, identified in September 2002, from April 2004, to June 30, 2005

b. Findings

No findings of significance were identified.

.2 Makeup and Purification System

a. Inspection Scope

The inspectors verified the licensee's handling of performance issues associated with the Makeup and Purification System, including the support systems for important components. This inspection consisted of evaluating the following specific licensee activities:

- work scheduling practices including consideration of risk of transient initiation while performing work on operating components;
- use of the condition report process and work order notification system in identifying deficiencies and issues with the equipment;
- that maintenance activities on the components had been assigned appropriate risk classification;
- that goals and corrective actions for the long term reliability were appropriate;
- that short term corrective actions were appropriate for deficiencies with potential for significant operator workarounds or potential for pump trips; and
- that maintenance rule system status determination appeared appropriate for the equipment's recent history and current open work items.

Additionally the inspectors walked down the components while operating and shutdown, to evaluate the adequacy of past repair activities.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

.1 Repair of Makeup Water Valves MU362 and MU363

a. Inspection Scope

The inspectors reviewed the risk impact, the maintenance activities, and the operator compensatory actions associated with the maintenance and repairs of makeup water valves MU362 and MU363. This maintenance activity was chosen as having elevated risk impact based on the licensee's reliance on an emergency boration flowpath to the reactor coolant system, since the clearance required by the valve maintenance isolated the normal boration flowpath from both boric acid addition tanks.

b. Findings

Introduction: A self-revealing Non-Cited Violation of Technical Specification (TS) 6.8.1.a, having very low safety significance, was identified when the control room staff attempted to add water to the makeup tank using equipment that had been removed from service as part of the clearance which supported work on makeup valves MU362 and MU363. Procedure DB-OP-0000, "Conduct of Operations," Revision 10, required, in part, that Operations personnel "shall be responsible for reviewing, completing, and understanding the turnover checklists applicable to their shift position prior to assuming the shift." The control room staff was unaware of the status of the normal makeup water sources to the reactor coolant system, even though the system's status was clearly documented in the Limiting Condition for Operation Tracking Log, a document which was required to be reviewed by the Shift Manager, the Unit Supervisor, and the Reactor Operator prior to shift turnover.

Description: On August 18, 2004, the night shift crew reviewed and established a clearance to support maintenance activities on makeup system valves MU362 and MU363. It was identified by the night shift operations personnel that the clearance impacted the ability to utilize the normal boration flowpath to the reactor coolant system. To compensate for taking this flowpath out of service, the licensee decided that, if necessary, the emergency boration flowpath would be used in lieu of the normal boration flowpath and a specific operator was assigned to perform this system lineup. Additionally, the personnel reviewing the clearance identified that the water flowpath from the clean waste system and demineralized water system would also be made unavailable. This information was documented in the licensee's Limiting Condition for Operation Tracking Log.

Subsequent to shift turnover, the day shift reactor operator attempted to add water to the makeup tank. While attempting to establish the proper flowpath, the operator observed an unexpected indication of flow when MU40 was opened. Valve MU40 was immediately closed. An investigation to determine the source of the unexpected flow revealed that the source of the back leakage was flow through a partially open drain valve. This investigation also made the control room staff aware that the clearance for the MU362 and MU363 maintenance also isolated the water flowpath from the clean waste system and demineralized water system.

Although there were many missed opportunities to adequately communicate the impact of the clearance that was put in place to support the MU362 and MU363 maintenance, this information was accurately documented in the Limiting Condition for Operation Tracking Log. The inspectors reviewed administrative procedure DB-OP-00100, "Shift Turnover," Revision 10, and its associated turnover checklists, and noted that each of the licensed operators [Shift Manager, Field Supervisor, Unit Supervisor, and Reactor Operator] are required to review the Limiting Condition for Operation Tracking Log prior to assuming the shift. Since the control room operators did not know the status of the normal makeup flow path, this portion of their turnover was inadequate.

Analysis: The inspectors determined that a performance deficiency existed because operations personnel failed to meet a requirement to maintain a thorough knowledge and understanding of the operation of all systems and equipment in their assigned areas described in DB-OP-00000, "Conduct of Operations," Revision 10, Section 6.2.1. Since there was a performance deficiency, the inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Dispositioning Screening," of IMC 0612, "Power Reactor Inspection Reports." The inspectors concluded since the operator's lack of knowledge of system status challenged their ability to adjust control rod index by adding water to the reactor coolant system and to perform selected abnormal operating procedures prepared to address small reactor coolant system leaks, the issue was more than minor. Specifically, it: (1) involved the human performance attribute of the Initiating Events Cornerstone; and (2) affected the 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. This finding was of very low safety significance because, during the time period the clearance impacted the availability of the makeup water sources, neither the ability to control makeup tank water level or to maintain an appropriate rod control index were challenged. Therefore, the finding was considered to be of very low safety significance (Green).

Enforcement: Technical Specification 6.8.1.a. and Regulatory Guide 1.33, 1978 Section 1.a, "Authorities and Responsibilities for Safe Operation and Shutdown," required that activities associated with safe operation and shutdown of the plant be performed in accordance with written procedures or documented instructions appropriate to the circumstances. DB-OP-00000, "Conduct of Operations," Revision 10, Section 6.2.1., required operators to maintain a thorough knowledge and understanding of the operation of all systems and equipment in their assigned areas. Contrary to the above, on August 18, 2004, the reactor operator failed to maintain a thorough knowledge and understanding of the operation of all systems and equipment in their assigned areas. Specifically, he attempted to add water to the makeup tank utilizing equipment that was out of service as part of a clearance that was supporting unrelated maintenance. Because this violation was of very low safety significance and because it was entered into the corrective action program, the NRC treated this issue as an Non-Cited Violation (05000346/2004014-01), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. This issue was entered into the licensee's corrective action program as CR 04-05174.

.2 ARTS Channel 1 Deficiency Identified During SFRCS Logic Channel 1 Logic Channel Testing

a. Inspection Scope

On September 28, 2004, the licensee was performing functional surveillance testing of SFRCS logic channel 1. Part of this test verified that anticipatory reactor trip system (ARTS) channel 1 received and processed through its logic a SFRCS actuated signal. To ensure that other channels are not impacted by this testing, a test trip bypass switch for ARTS channel 1 was placed in the "SFRCS" position. In this position, the signal received by the ARTS logic channel 1 will not be processed to the other 3 ARTS channels. When the test signal was generated in SFRCS channel 1, ARTS channel 1 did not indicate that the signal was received and processed. The inspectors reviewed the licensee's response to the failure which included declaring ARTS channel 1 inoperable and tripping reactor trip breaker B (the trip breaker associated with ARTS channel 1) and determining that there was reasonable assurance that SFRCS channel 1 was operable. The inspectors also observed the licensee's actions in discussing the issue and observed a portion of the field troubleshooting activities. The inspectors reviewed the licensee's actions associated with evaluating the impact of the ARTS failure on their risk profile and the associated re-scheduling of work activities already in the work schedule. Additionally, the inspectors observed corrective action work activities and verified that the work being performed was consistent with the established schedule and associated risk profile.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected condition reports which discussed potential operability issues for risk significant components or systems. These condition reports and applicable licensee operability evaluations were reviewed to determine whether the operability of the components or systems was supported. The inspectors compared the operability and design criteria in the appropriate sections of the USAR to the licensee's evaluation of the issues to verify that the components or systems were operable. Where compensatory measures were necessary to maintain operability, the inspectors verified that the compensatory measures were in place, would work as intended, and were properly controlled.

The two samples evaluated were:

- Operability Evaluation 04-0020 for CR 04-04737 [Fire Protection Piping Operability Concern] which reviewed the effect of piping pressure increases due to heating up of water in the closed system of fire protection piping and nozzles servicing the component cooling water pump and heat exchanger room; and
- Operability Evaluation 04-0023 for CR 04-05594 [Station Battery Rack Bolting] which evaluated the adequacy of bolting material used in the construction of the station battery racks.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed licensee actions to work around an issue with ICS that caused unexpected reactor coolant system average temperature swings. During this inspection period, the plant experienced a deficiency associated with the monitoring of an individual control rod position in relation to the rod's relative demanded position. To do this monitoring, control room operators must position a toggle switch which changes indication between actual and relative. Because of indication relay problems with control rod 2-3, to complete the comparison between actual and relative for all rods, several cycles of the toggle switch could be required The toggling of the switch introduced electrical spikes into circuitry feeding logic gates that, on August 25, 2004, caused the Diamond Panel (rod control system) to automatically switch to manual and the integrated control system to switch to track. In this condition, reactor coolant average temperature control is through feedwater demand changes. During this inspection period, the temperature deviation output signal developed within the ICS system caused excessive feedwater demand signals which forced control room operators to take manual control of feedwater demand. To perform monitoring of rod position while minimizing the possibility of temperature and feedwater transients, the licensee developed procedures to work around the problem.

The inspectors reviewed standing orders developed to place feedwater demand in manual, along with rod control and ICS reactor demand in manual, prior to performing the surveillance testing on control rod position. The inspectors observed the actions necessary to prepare for the surveillance, the performance of the surveillance test, and the actions necessary to return the ICS to automatic operation. Additionally, the inspectors evaluated how these actions might impact the operator's ability to perform other required duties. The inspectors verified that Operations personnel knew what action would be taken with ICS in the event of conditions that initiated a plant and turbine runback. The licensee took actions to correct the issues or to minimize the potential for a significant transient (see Section 1R17 and 4OA3).

b. Findings

No findings of significance were noted.

1R17 Permanent Modifications (71111.17A)

a. Inspection Scope

The inspectors evaluated Engineering Change Package 04-0331-00; "Rod Control Automatic Transfer to Manual," as a sample of a permanent plant modification.

The inspectors reviewed the modification prior to installation and testing to verify that the design basis, licensing basis, and performance capability of the rod control system was not degraded by the on-line installation of the modification and specifically that the modification did not adversely impact the transfer of rod control from automatic to manual if an out-of-sequence rod was detected. The inspectors evaluated the adequacy of the design of the modification by performing a review of the modification's impact on signals that would be generated under rod position switching conditions and upon detection of a valid rod out of sequence signal. The inspectors reviewed specified rod control and integrated control system testing conditions for the impact on the ability of the control room staff to respond to plant events, including emergency or abnormal conditions. Additionally, the inspectors witnessed the modification being accomplished on-line and the post modification on-line testing to verify proper operation of the modified system.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance testing activities to ensure that the testing adequately verified system operability and functional capability with consideration of the actual maintenance performed. The inspectors used the appropriate sections of the TSs and the USAR, as well as the documents listed at the end of this report, to evaluate the scope of the maintenance and verify that the work control documents required sufficient post-maintenance testing to adequately demonstrate that the maintenance was successful and that operability was restored. The inspectors observed and evaluated test activities associated with the following five samples:

- replacement of ICS relay module 3-2-1, summer plus integral module 4-7-2 and operational test on August 26, 2004, to verify proper feedwater demand signals with reactor coolant average temperature regulating demand;
- pressurizing and filling turbine generator electro-hydraulic control accumulators 1 and 2 while the turbine generator was on-line on September 15, 2004, and September 17, 2004, after installation of a replacement for accumulator 2;
- voltage measurements, test trip bypass switch cycling, and functional testing during restoration of Anticipatory Reactor Trip System channel 2 on September 17, 2004, after the channel de-energized unexpectedly due to an momentary internal short circuit that caused the power supply breaker to open;
- restoration and flow testing of decay heat pump 2 on September 23, 2004, after replacement of oil in a bearing, replacement of pump mechanical seal injection cyclone separators, and work involving testing and cycling a valve in the suction lineup from the borated water storage tank; and
- cleaning and inspecting the motor for valve SW1367 [CAC 1-2 Inlet Isolation Valve] on August 19, 2004.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

On September 24, 2004, the inspectors observed the collection of the daily Reactor Coolant System sample from the Purification Letdown Demineralizer inlet sample point and reviewed test results to verify that the Reactor Coolant System specific activity was within TS limits. The activity was selected based on its importance in verifying the barrier integrity of the RCS. The inspectors used the documents listed at the end of this report to verify that the test met the TS frequency requirements; that the test was conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; and that the test acceptance criteria were met.

b. Findings

No findings of significance were identified

EP6 Drill Evaluation (71114.06)

a. Inspection Scope:

The inspectors monitored the licensee's emergency preparedness exercises conducted on September 16, 2004, from various locations and perspectives. The observations included licensee preparations, evaluation of drill conduct, review of the drill critiques, and identification of weaknesses and deficiencies. The inspector reviewed the licensee's scenario and preparations. The inspectors specifically observed drill activities and personnel performance in the simulator control room, the technical support center, and the emergency operating facility. The inspectors noted the communications, accuracy of situation evaluations, and reporting (simulated) to appropriate agencies. Finally, the inspectors reviewed the licensee's drill critique to assure that weaknesses and deficiencies were acknowledged and appropriateness of corrective actions identified.

b. Findings:

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors reviewed the reported data for the following Performance Indicators:

- Reactor Coolant System Leakage (January 2004 to June 2004); and
- Reactor Coolant System Specific Activity (April 2004 to June 2004).

The inspectors used the definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 2. The inspectors reviewed station logs, condition reports, and daily chemistry and reactor coolant leakage data to verify the accuracy of the licensee's data submission.

Due to the length of the Davis-Besse extended outage, limited data was available to support both of these performance indicators. To compensate for the limited data available for certain Performance Indicators, the Davis-Besse 0350 Oversight Panel approved an inspection strategy which supplemented the baseline Performance Indicator inspections with additional baseline inspection activities which supported the validity of the Performance Indicator data. These additional inspection activities were as follows:

- on two occasions [Davis-Besse Inspection Reports 2004-06 and 2004-07] , inspectors observed the performance of surveillance procedure DB-SP-O3557, “Reactor Coolant System Water Inventory Balance”; and
- on two occasions [Davis-Besse Inspection Reports 2004-08 and 2004-14], the inspectors observed the performance of radiochemistry procedure DB-CH-01816, “Gross Specific Activity.”

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Review

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment deficiencies or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s corrective action program. This screening was accomplished by reviewing documents entered into the licensee corrective action program and review of document packages prepared for the licensee’s daily Management Alignment and Ownership Meetings.

b. Findings

No findings of significance were identified.

.2 Operator Performance Collective Significance Review

a. Inspection Scope

The inspectors selected for detailed review CR 04-04425 which was established by the licensee to perform a collective significance review for various licensee errors associated with TS compliance. The initiating event for the collective significance review was the licensee’s identification, on July 6, 2004, that on multiple occasions, proper TS specified channel checks of RCS flow channels associated with the reactor protection system (CR 04-04406) did not occur. Additionally, the licensee identified four similar issues that had occurred since January 2004, that were previously reviewed and evaluated by the licensee. Three of those previous events involved root cause determinations and all involved the development of corrective actions. The inspectors reviewed the actions developed as a result of the collective significance review and also reviewed the previously developed corrective actions and the status of implementation of those actions. Additionally, the inspectors reviewed the relationship of this collective significance review to that of an earlier collective significance review associated with similar events.

b. Findings

Introduction: The inspectors identified a Non-Cited Violation of 10CFR 50, Appendix B, Criterion V, having very low safety significance, for the licensee's failure to properly implement procedure NOP-LP-2001, "Condition Report Process" and identify corrective action to prevent repetition of conditions adverse to quality. Specifically, subsequent to repetitive TS violations and licensed operator performance errors, implemented corrective actions were not effective in precluding recurrence of similar events. Additionally, during this time period, effectiveness reviews, that were specified in the licensee's corrective action program to gauge the effectiveness of corrective actions implemented to prevent the repetition of significant conditions adverse to quality, were deferred.

Description: Procedure NOP-LP-2001, "Condition Report Process," is a quality related administrative procedure. This procedure specifies, in part, that:

- the condition report owner is responsible for ensuring adequate and effective corrective actions are developed to resolve the issue identified in the CR;
- corrective action is a general term to describe any measure taken to address a condition or its causes and includes compensatory measures, remedial actions, preventive actions, enforcement actions and other actions;
- all condition reports requiring a root cause analysis shall include preventive actions for identified root causes;
- corrective actions developed or implemented for conditions reports for root cause evaluations are intended to preclude repetition; and
- effectiveness reviews are mandatory for all condition reports performing a root cause evaluation.

The events described in CR 04-04425 are similar to events that were reviewed and discussed in IR 50-340/03-011 (DRP). In that report, the NRC's Restart Readiness Assessment Team discussed ineffective corrective action following several operational events which occurred during and after the September 2003, Mode 3 normal operating pressure and temperature test. These events were documented in the licensee's corrective action program as CR 03-08418 (Operations Events - Collective Significance Review; October 1, 2003). A collective significance review associated with CR 03-08418 considered the following condition reports:

- CR 03-07746 (Core Flood Valve 1B Inadvertent Opening; September 15, 2003);
- CR 03-07710 (Steam Generator Overfill; September 13, 2003);
- CR 03-07262 (Auxiliary Feed Water Check Valve Test Procedure Use; September 3, 2003);
- CR 03-07930 (TPCW Spill; September 22, 2003);
- CR 03-07689 (RC4610A Testing; September 12, 2003); and
- CR 03-08374 (Reactor Trip on Shutdown High Pressure; September 30, 2003).

On July 6, 2004, the licensee initiated a collective significance review (CR 04-04425) for various TS related events that were the result of Operations department errors which had occurred since January 2004. The specific events considered were:

- CR 04-00181 (Missed TS Action Statement; January 6, 2004);
- CR 04-01230 (Failure to Recognize TS Entry for Radiation Elements 4597; February 12, 2004);
- CR 04-02767 (Failure to Perform Re-Stroke of AFW Valve Following Stroke Time Exceeding Expected Data; April 19, 2004);
- CR 04-03800 (Failure to Perform the Correct Electrical Alignment Surveillance for Operational Mode 1; June 7, 2004); and
- CR 04-04406 (Failure to Perform a Proper TS Surveillance for RCS Flow; July 6, 2004).

As required by NOP-LP-2001, corrective actions, which included preventive actions where applicable, were developed.

The significance review associated with CR 03-08418 identified 73 corrective actions. All but 2 of those corrective actions were completed by May 27, 2004, with a significant number being completed in 2003. Additionally, an effectiveness review was completed and a report provided on the results of that review. The plant's senior leadership team concurred with the report on January 21, 2004. The report concluded that the operations improvement implementation action plan, which addressed the events listed, was somewhat effective in improving operator performance and that additional management attention was needed to ensure continued improvement in operator performance.

CR 04-0181, CR 04-01230, CR 04-03800, and CR 04-04406, which were evaluated as part of the corrective actions associated with CR 04-04425, had root cause analyses performed. Cause codes identified in the root cause reports included "management expectations not communicated or worker accepted," "self-checking not applied to ensure correct component/train," and "job scoping or walkdown incomplete." Effectiveness reviews were scheduled to evaluate the effectiveness of the developed corrective actions, as specified by NOP-LP-2001. The corrective actions associated with CR 04-00181, CR 04-01230, and CR 04-03800 were substantially complete at the end of this inspection period, but the corrective action effectiveness reviews were scheduled for 2005.

Corrective actions associated with CR 04-04425 and CR 04-04406, were scheduled to be completed in the fourth quarter of 2004 or in 2005. The corrective actions associated with the collective significance review of CR 04-04425 were long-term corrective actions which include training in performance management. Additionally, Shift Manager roles and responsibilities that are delegated, were scheduled to be reviewed and clarified, with any needed process changes. These two items were scheduled to be completed by January 5, 2005.

The inspectors noted that operational problems continued to occur even after implementation of numerous preventive actions. Specifically, all of the conditions reports considered by CR 04-04425 occurred after implementation of corrective actions associated with CR 03-08418 and completion of its effectiveness review. Events associated with CR 03800 and CR 04-04406 occurred after reported implementation of preventive actions for CR 04-00181. Additionally, CR 04-05174 (Boron Injection

Flowpath Clearance Issues) documented an operational problem with the status of the boric acid transfer system that occurred on August 18, 2004, after implementation of preventive action for CR 04-00181 and CR 04-01230 (Section 1R13). Other operational errors have included:

- CR 04-05231 (High Pressure Feedwater Heater Level Transient During Tagout Restoration; August 20, 2004); and
- CR 04-05256 (Flow Set Up Incorrectly for Motor Driven Feedwater Pump Test; August 24, 2004).

Analysis: The inspectors determined that ineffective corrective actions resulting in continuing issues with errors associated with TS surveillances and equipment operation was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspections Reports," Appendix B, "Issue Disposition Screening," issued on June 20, 2003, because if left uncorrected the issue would become a more significant safety concern.

The inspectors determined that the finding could not be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," because this issue did not directly affect any of the cornerstone attributes in accordance with IMC 0612 Appendix B. Therefore, this finding was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c, and determined to be of very low safety significance because no safety systems were degraded nor was any safety equipment rendered inoperable directly due to this issue.

Enforcement: 10 CFR 50, Appendix B, Criterion V, required that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances and shall be accomplished with those instructions or procedures. Procedure NOP-LP-2001, "Condition Report Process," was a quality related administrative procedure that required development of corrective actions for root cause evaluations which were intended to preclude repetition of adverse conditions. Contrary to those requirements, preventive actions implemented for the root cause analyzes associated with CR 04-00181 and CR 04-01230 were ineffective in precluding recurrence of the adverse conditions. Because the violation was of very low safety significance and the events were entered into the licensee's corrective action program (CR 04-06759), this violation was being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000346/2004014-02).

4OA3 Event Followup (71153)

.1 Control Rod Drive System and Integrated Control System Problems

On August 25, 2004, while performing a routine verification of control rod absolute and relative position indication, the Diamond Panel (rod control panel) automatically transferred to manual as soon as the position indication toggle switch was moved from the "absolute position" to the "relative position." On August 26, 2004, as part of troubleshooting efforts to identify the source of problem, the operators placed the Diamond Panel in manual. At this point, ICS feedwater loop demand signals ramped up, which resulted in increased feedwater flow to the steam generators. The operators quickly placed both feedwater loop demands in manual and restored the appropriate feedwater flows. This feedwater transient resulted in an increase in reactor power to approximately 102 percent. Power was promptly reduced to below 100 percent power.

Through troubleshooting efforts, the licensee identified an auxiliary relay module and a summer plus integral module, associated with Tave signal [used to modify feedwater] as the most probable cause of the ICS feedwater demand transient. These modules were replaced on August 26, 2004, and post maintenance testing verified their proper operation.

On August 27, 2004, the licensee transferred the Diamond Panel to manual, as a compensatory measure to address the issue discussed in the first paragraph, prior to performing the routine once per shift API/RPI verification surveillance. At this time, the feedwater loop demand malfunction repeated. The operators had briefed this as a contingency during their pre-evolution brief and acted promptly to correct the demand by placing the feedwater demand in manual. During this transient, reactor power increased approximately 0.4 percent.

The licensee initially determined that the cause of the August 27, 2004, transient was due to the malfunction of the same modules that were replaced on August 26, 2004. The licensee replaced the modules with another available set and reduced module amplifier gain to slow down any potential future transients. The licensee, to gain a better understanding of the failure mechanism, mocked up in the workshop the applicable portions of ICS but was unable to reproduce, with the replaced modules, the transient observed in the plant. On August 28, 2004, the licensee issued standing orders to place feedwater demand in hand prior to placing the Diamond panel in manual. Subsequently, those orders were withdrawn after installation of a modification that eliminated the unwanted transfer of rod control to manual when switching between absolute rod position indication and relative rod position (see Section 1R17).

.2 <u>(Closed) LER 50-346/03-014-00</u>: Steam Feedwater Rupture Controls System Re-Energizes in a Blocked Condition.

a. <u>Inspection Scope</u>

The inspectors reviewed the LER and CR 03-08917. The CR documented the finding of the condition, the probable cause, and the potential system response in the event of a faulted steam generator concurrent with a loss of offsite power. Additionally, the inspectors reviewed the USAR sections that the licensee stated addressed an accident scenario that mimicked the conditions that would occur should a faulted steam generator occur concurrent with a loss of offsite power and with Steam Feedwater Rupture Controls System (SFRCS) logic channel 4 re-energizing in a blocked condition.

b. Findings

Introduction: A Green self-revealing NCV was identified for failure to comply with the requirements of TS 3.3.2.2.

Description: On October 15, 2003, while in mode 5, the licensee was accomplishing planned work that included de-energizing SFRCS. Upon re-energization of SFRCS logic channel 1, the channel re-energized in a blocked condition which was not expected by licensee personnel. Subsequent testing confirmed that all 4 of the logic channels in the SFRCS were susceptible to re-energizing in the blocked state. Two of the 4 logic channels are subject to de-energization and re-energization during a loss of offsite power (LOOP). The block function is available to inhibit steam generator isolation during anticipated low pressure conditions that would be experienced in plant startup and shutdown. Initiation of a blocked state during plant operation could inhibit faulted steam generator isolation in the event of a faulted steam generator concurrent with a LOOP. Subsequent analysis by the licensee found that only logic channel 4 was of concern with steam generator number 2 experiencing a fault concurrent with a LOOP. This resulting condition is similar to that analyzed in USAR Section 3.6.2.7.1.6. That section reviews the consequences of mass and energy releases due to the postulated delivery of auxiliary feedwater to an assumed faulted number 2 steam generator for 10 minutes at which time operator action is taken to stop the auxiliary feedwater flow.

Further investigation by the licensee determined that the original design of SFRCS became inadequate when the system was modified in 1987 and 1988 to change the original auctioneered power supply scheme to a design in which interruptible power supplies were provided for SFRCS logic channels 3 and 4. That modification package did not identify adequate testing requirements for ensuring that blocking was not initiated upon energization of the logic channels. Technical Specification 3.3.2.2 requires that SFRCS shall be operable in Modes 1, 2, and 3. The plant has operated in Mode 1, 2, and 3 since 1988 and prior to the discovery of the design inadequacy described in the LER.

Licensee investigations, subsequent to the October 2003, event, concluded that at least two procedures, as originally written, recognized that the logic channels could return in either a blocked or unblocked state upon re-energization. Also, a licensee review of previous test results concluded that there were at least two times, between 1988 and October 2003, where SFRCS logic channels re-energized in a blocked condition. The licensee did not realize the potential consequences of this condition until they reviewed the October 15, 2003, event. The licensee initiated a design change to correct the condition. The modification was reviewed in IR 50-346/2003-025.

Analysis: The inspectors determined that failing to properly review and test modifications for impacts on the proper operation of SFRCS was a performance deficiency warranting a significance determination. The inspectors concluded that the finding was greater than minor, in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on June 20, 2003. The finding involved the attributes of design control and equipment reliability and could have affected the mitigating systems objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors completed a phase 1 significance review in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," issued on September 10, 2004, and determined that the finding was of very low safety significance. The finding did not result in a loss of safety function since the starting of the auxiliary feedwater system was not affected and a faulted number 2 generator could be isolated with operator action even with a LOOP that included SFRCS channel 4 re-energizing in a blocked condition. Additionally, feeding of a faulted number 2 steam generator with auxiliary feedwater for 10 minutes was a condition analyzed in the USAR.

Enforcement: The LER describes plant operation in a condition prohibited by TS 3.3.2.2. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (CR 03-08917) and was addressed in an LER, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000346/2004014-03). This LER is closed.

.3 (Closed) LER 50-346/99-003-01: Failure to Perform Engineering Evaluation for Pressurizer Cooldown Rate Exceeding TS Limit.

On August 26, 1999, the licensee submitted LER 99-003-00 for the failure to perform an engineering evaluation, prior to unit restart from a mid-cycle outage, for a pressurizer cooldown rate that exceeded the TS limit of 100 degrees per hour. That LER was reviewed and closed in IR 50-346/99-010 dated October 8, 1999. In that LER the licensee reported that there were no LERs, within the last 3 years of the event, that related to overcooling the reactor coolant system or the pressurizer. On July 15, 2003, the licensee provided a letter that stated that, contrary to statement in revision 0 of the LER, LER 98-011 reported an overcooling of the reactor coolant system, although the cooldown rate did not exceed TS limits. Inspection Report 50-346/03-19 (DRP) documented the NRC's review of the issue and determination that it was a violation of minor significance. On February 3, 2004, the licensee completed revision 01 to LER 99-003 which stated that, during the 3 years prior to the 1999 event, there were no LERs documenting violations of TS cooldown limits. The inspectors did not identify any additional items of significance during the review of this LER revision. This LER is closed.

.4 (Closed) LER 50-346/02-005-00 through -02; VIO 50-346/03-015-05: Potential Clogging of the Emergency Sump Due to Debris in Containment.

This LER documented discovery of a 3/4-inch wide by 6-inches long gap in the containment emergency sump screen that was larger than allowed by design basis and unqualified coatings and other debris in containment that could potentially have blocked the containment emergency sump screen. The licensee determined that the gap had existed since original construction. NRC evaluation of the unqualified coatings and debris was discussed in NRC Inspection Report No. 50-346/03-15, issued on July 30, 2003. On October 7, 2003, the NRC issued the yellow final significance determination, which included a Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions." Analysis of the impact of the gap on the function of the containment emergency sump is discussed below.

The performance deficiency for the gap issue was the licensee's failure to correctly verify the adequacy of the design of the containment emergency sump screen, and to implement effective corrective actions to repair the gap. Because the licensee had several opportunities to correct the degraded condition of the sump, the inspectors focused on the corrective action aspects of this performance deficiency.

Analysis: In accordance with IMC 0612, Appendix B, "Issue Screening," the inspectors determined that the existing gap in the containment emergency sump screen was more than minor safety significance because the deficiency is associated with the objective and attributes of the Mitigating Systems and the Barrier Cornerstones. Specifically, the containment emergency sump screen was sized to pass no more than 1/4-inch debris particles and debris larger than 1/4-inch could have potentially damaged emergency core cooling system (ECCS) equipment and/or clogged the containment spray system (CSS) nozzles.

In accordance with IMC 0609, Appendix A, Attachment 1, the inspectors performed a SDP Phase 1 screening and determined that the finding potentially affected the Mitigation Systems and Barrier Cornerstones. A Phase 2 analysis was required because two cornerstones were impacted. The decision to enter the Phase 2 process was conservative because it was not determined by a specific transport analysis that the gap would actually result in a loss of equipment (e.g., high and low head safety injection, containment spray nozzles) due to normal containment sump debris pass through. This analysis does not consider any sump debris (e.g., unqualified containment coatings) other than would be expected during normal sump recirculation. This approach was used because the performance deficiency that caused the sump screen gap is not sufficiently linked to the performance deficiency that caused the unqualified coatings mentioned above.

In the Phase 2 analysis, the inspectors evaluated potentially affected equipment by reviewing Table 2 of the Davis-Besse Risk Informed Inspection Notebook. The affected functions for accident mitigation were both high pressure recirculation (HPR) and low pressure sump recirculation (LPR). The equipment of interest were the high and low pressure safety injection pumps and associated equipment needed to move sump inventory to the reactor pressure vessel during certain accident sequences requiring sump recirculation. In selecting the worksheets to be evaluated in the Phase 2 SDP, the inspectors evaluated the appropriate initiating events that would result in the transport of containment material to the sump and potentially damage the downstream equipment mentioned above.

According to research from GSI-191, "The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," most of the debris that is transported to the containment emergency sump is fibrous, with the amount of fiber passing through the containment emergency sump screen being proportional to the flow area. The gap provides only a very slight increase in flow area. In addition, insulation and other debris would have to arrive at the exact point of the containment emergency sump screen gap with the correct orientation for pass through. If debris were to enter into the ECCS system, it could either cause a pump to fail or clog the flow path. Most sections of the ECCS piping are large in comparison to the debris so the debris could accumulate in a low flow area or on valve internals with no measurable effect on ECCS flow.

In the unlikely event that debris reached the low pressure safety pumps, the pumps would most likely pass the debris without a failure because the debris would likely be broken into small pieces by the low pressure injection pumps. The debris would then have to pass through the low pressure area of the system to the high pressure injection pumps.

Damage to the high pressure injection pumps would also be unlikely. As part of NRC's evaluation of a previous high pressure injection pump design issue discussed in NRC Inspection Report 50-346/04-05, high pressure injection pump capability for HPR was analyzed. The NRC determined that initiation of HPR following a loss of feedwater event or a loss of coolant from reactor coolant pump seals were the most limiting scenarios for containment debris generation and transport. The NRC concluded that the debris generated would be insufficient to render the high pressure injection pumps incapable of performing their safety function. A significant contributor to the White finding in Report 50-346/04-05 was the risk increase due to internal fires and conservative assumptions that are not applicable to this gap issue.

Based on the preceding qualitative analysis, the inspectors concluded that the gap in the containment emergency sump screen would not likely affect mitigation for any of the initiating events defined in Table 2 of the Davis-Besse Risk Informed Inspection Notebook. Therefore, the issue is of very low safety significance (Green). In order to confirm this conclusion, the inspectors addressed the SDP Phase 1 screening questions that were bypassed initially in this evaluation. The Phase 1 results for the Mitigating Systems Cornerstone are as follows. The sump screen gap: 1) was not a design or qualification deficiency resulting in a loss of function per Generic Letter 91-18, Revision 1; 2) did not represent an actual loss of safety function of a system; 3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; 4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours; and 5) did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event. For the Barrier Cornerstone, the inspectors evaluated the effects of potentially clogging the containment spray nozzles and determined, using guidance in IMC 609, Appendix H, that containment spray is not a large early release frequency contributor (see Table 4.1 of Appendix H).

Enforcement: This failure to comply with 10 CFR Part 50, Appendix B, Criterion XVI, is of very low safety significance and the violation has been adequately resolved with installation of a new containment emergency sump. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000346/2004014-04). The NRC inspected the new containment emergency sump, which is documented in Inspection Report 50-346/03-06.

The NRC concluded that the containment emergency sump design modification was consistent with the design and licensing basis requirements, and based on field walkdowns the modification installation was adequately implemented consistent with the design.

This issue of the containment sump was also contained in Restart Checklist Item 2.c.1, "Emergency Core Cooling System and Containment Spray System Sump." This Restart Checklist Item was closed in NRC inspection Report 50-346/03-17.

4OA4 Cross-Cutting Aspects of Findings

Section 4OA2 describes the licensee deferred corrective action effectiveness reviews while experiencing continuing issues with TS issues and equipment operation. The inspectors determined that the non-recognition of a continuing trend despite actions taken to prevent similar events, besides directly impacting problem identification and review, also affected the cross-cutting area of Human Performance. Preparers of and those approving deferral actions for corrective action effectiveness reviews had information from ongoing activities that issues previously addressed were not corrected and still prepared and approved the deferral requests.

4OA5 Other Activities (93812)

Following restart authorization, Inspection Procedure 93812 remained in effect to facilitate the documentation of issues not specifically covered by existing procedures, but are important to the evaluation of the licensee’s performance post-restart. This inspection procedure remains in effect as part of the integrated resident inspection report until a time to be determined by the Davis-Besse Oversight Panel.

.1 Review of Licensee Performance During Emergent Equipment Issues

During this inspection period the licensee experienced various equipment issues which cumulatively would be more significant than if considered by themselves. These issues included issues with a ground on turbine EHC, problems with a card that permitted transfer from the control room of control rods to the auxiliary power supply, indication of a small turbine condenser tube leak, control rod drive system issues that caused the integrated control system to transfer to track when switching between absolute position indication and relative position indication, and unexpected high feedwater demand signals when transferring ICS out of track to full automatic operation. The inspectors, in addition to reviewing directly many of the activities associated with each issue, also observed management actions associated with assessing the cumulative effects of the issues and in management of priorities. The inspectors determined that the licensee was effective in addressing these emergent maintenance issues.

4OA6 Meetings

Exit Meeting

The inspectors presented the inspection results to Mr. M. Bezilla, and other members of licensee management on September 29, 2004. The licensee acknowledged the findings presented. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

B. Allen, Director, Plant Operation
M. Bezilla, Site Vice President
L. Harder, Manager, Radiation Protection
D. Kline, Manager, Security
S. Loehlein, Director, Station Engineering
L. Myers, Chief Operating Officer, FENOC
K. Ostrowski, Manager, Plant Operations
C. Price, Manager, Regulatory Compliance
M. Stevens, Manager, Maintenance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000346/2004014-01	NCV	Reactor Operator Attempted to Add Water to the Make Up Tank Utilizing Equipment That Was Out of Service
05000346/2004014-02	NCV	Failure to meet procedure implementation requirements of 10 CFR 50, Appendix B, Criterion V, Associated with Recurring Operations Performance Issues
05000346/2004014-03	NCV	Failure to Comply with TS Because of Design Inadequacy in the Steam Feedwater Rupture Controls System
05000346/2004014-04	NCV	Failure to Correctly Verify the Adequacy of the Design of the Containment Emergency Sump Screen, and to Implement Effective Corrective Actions to Repair a 3/4-Inch Wide by 6-Inches Long Gap

Closed

50-346/99-003-01	LER	Failure to Perform Engineering Evaluation for Pressurizer Cooldown Rate Exceeding TS Limit
50-346/03-014-00	LER	Steam Feedwater Rupture Controls System Re-Energizes in a Blocked Condition
50-346/02-005-00 to 02	LER	Potential Clogging of the Emergency Sump Due to Debris in Containment
50-346/03-015-05	VIO	Potential Clogging of the Emergency Sump Due to Debris in Containment

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless stated in the body of the inspection report.

1R04 Equipment Alignment

DB-OP-06012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 18
Drawing OS-004, Sheet 1; Decay Heat Removal/LPI System; Revision 37
DB-OP-06334; Station Blackout Diesel Generator Operating Procedure; Revision 07
Operational Schematic OS-041D; Station Blackout Diesel Generator Lube Oil and Jacket Water; Revision 10
Operational Schematic OS-041E; Station Blackout Diesel Generator Air Start/Engine Air System; Revision 10
Operational Schematic OS-041F; Station Blackout Diesel Generator Electrical Control and Fuel Oil System; Revision 02
DB-OP-06233; Auxiliary Feedwater System; Revision 07
Operational Schematic OS-017A, Sheet 1; Auxiliary Feedwater System; Revision 20
Operational Schematic OS-017B, Sheet 1; Auxiliary Feedwater Pumps and Turbines; Revision 23

1R05 Fire Protection

Davis-Besse Nuclear Power Station Fire Hazard Analysis Report
PFP-AB-328; Pre-Fire Plan, Component Cooling Water Heat Exchanger and Pump Room; Revision 03
Drawing A-223F; Fire Protection General Floor Plan El 585'; Revision 016

1R11 Licensed Operator Requalification Program

Applicable Drill Simulator Guide for the Observed Scenario
DB-OP-0000; Conduct of Operations; Revision 10
DBBP-OPS-0001; Operations Expectation and Standards; Revision 04
DBBP-TRAN-0017; Conduct of Simulator Training; Attachment 4; Crew Critique Form; Revision 00

1R12 Maintenance Effectiveness

CR 02-08570; SHRR: Repeat Issue - Relay Contacts in ICS
CR 02-09109; SHRR: ICS Design Issues
CR 02-09112; SHRR: Capacitors External to Modules
CR 04-05308; Plant Transient When Control Rods Placed in Manual
CR 03-10394; ICS DAAS Work Order Closed Without Followup Work Document

CR 04-05283; Rod Control Automatic Xfer to Manual When PI Panel Placed in Relative Position
CR-04-05445; Maintenance Rule (A)(1) Evaluation of the ICS/NNI System
CR 02-07051; SHRR - No PM for Calibrating ICS Modules
DB System Health Report, Panel 20, ICS/NNI; Second Quarter 2004
DB-PF-00003; Maintenance Rule; Revision 05
System Description SD-048; Makeup and Purification System; Revision 03
Operational Schematic OS-002; Sheet 1; Makeup and Purification System; Revision 21
Operational Schematic OS-002; Sheet 2; Makeup and Purification System; Revision 17
Operational Schematic OS-002; Sheet 3; Makeup and Purification System; Revision 27
Operational Schematic OS-002; Sheet 4; Makeup and Purification System; Revision 15
Operational Schematic OS-002; Sheet 2; Makeup and Purification System; Revision 21

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

Problem Solving Plan for ARTS 1/5 Light Did not Illuminate During SFRCS Testing (CR 04-05940); Revision 01
DB-MI-03211; Channel Functional Test of SFRCS Actuation Channel 1 Logic for Mode 1; Revision 09
SAP Order 200115322; DB-C5784A - Troubleshoot ARTS Channel 1
CR 04-05174; Boron Injection Flowpath Clearance Issues
Order 200002853 [MU362]
Order 200100750 [MU362]
DB-0492-0; Shift Manager Turnover Checklist
DB-0490-1; Reactor Operator Turnover Checklist
DB-0488-0; Unit Supervisor Turnover Checklist

1R15 Operability Evaluations

CR 04-04737; Fire Protection Piping Operability Concern
Drawing —357; Sprinkler System Room 328; Revision 1
Drawing M-016B; Station Fire Protection System; Revision 41
Plant Design Standard —602; Revision 30
CR 04-04510; CCW Heat Exchanger and Pump Room Sprinkler System
Davis-Besse Nuclear Power Station Fire Hazard Analysis Report
DB-OP-06601; Wet Pipe Sprinkler Systems; Revision 03
Drawing OS-047B, Sheet 1; Fire Suppression System; Revision 4
CR 04-05594; Station Battery Rack Bolting
Drawing E18-3; Layout for 60 Cell Battery Racks Heavy Seismic Resistant

1R16 Operator Workarounds

Standing Order 04-020; Interim Guidance for Operation of the Reactor Demand and Diamond Stations due to Problems with the Control Rod Drive System; August 28, 2004
Standing Order 04-021; Interim Guidance for Operation of Feedwater Loop Demand Hand/Auto Stations - FCI ICS 32B and FIC ICS 32A When Transferring the Reactor Demand and Diamond Stations to Manual; August 28, 2004
Operator Work Around Form for RPI/API Select Switch; August 31, 2004

White Paper - ICS Feedwater Control of RCS Tave During CRDs Troubleshooting; September 2, 2004

1R17 Permanent Modifications

ECR 04-0331-00; Rod Control Automatic Transfer to Manual; August 31, 2004
SAP Order 200110643; Monitor CRD Response Using DAAS
SAP Order 200111237; DB-C4801X: Implement ECR 04-0331-00
Problem Solving Plan for CR 04-05283; Rod Control Automatic Transfer to Manual when PI Panel Placed in Relative Position; Revision 1 dated August 27, 2004
CR 04-05509; Order Not Used for Soldering of Capacitor of CRD Output Driver Module

1R19 Post-Maintenance Testing

SAP Order 200110711; CR 04-05307 and 04-05308 ICS Transients
SAP Order 200110834; PS/DM ICS Transients
CR 04-05307; ICS Malfunction
CR 04-05308; Plant Transient When Control Rods Placed in Manual
Problem Solving Plan for Feedwater Control of Tavg Results in an Unexpected Correction to Total Feedwater Demand; August 28, 2004
Drawing OS-23, sheet 2; Turbine Electrohydraulic Control System; Revision 20
SAP Order 200096470; Accumulator T201-2 Leaking Nitrogen
DB-OP-06204; EHC System Operating Procedure; Revision 11
Drawing E-28; Anticipatory Reactor Trip System Logic Diagram; Revision 13
DB-MI-03058; RPS Channel 2 Calibration of Overpower, Power/Imbalance/Flow, and Power/Pumps Trip Function; Revision 15
DB-MI-03352; Channel Functional Test of PSL-4533B, 4534B and 4535B Main Feed Pump 1 and 2 Turbine Hydraulic Oil Trip and Main Turbine Oil Trip ARTS Channel 2; Revision 03
SAP Order 200113853; ARTS Ch2 Lost Power: Take Voltage Readings
CR 04-05696; Unexpected ARTS Ch. 2 Trip
DB-SP-03137; Decay Heat Train 2 Pump and Valve Test; Revision 10
DB-PF-06704, Attachment 6; Decay Heat Pump 2 DP vs. Flow IST Acceptance Criteria for Quarterly Surveillance Test; Revision 13
Drawing OS-004, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 37
SAP Order 200088498; MP42-2 - Flush 1B Bearing, leak-Pump 1B
SAP Order 200088892; DH Pump #2 Cyclone Separator - Clean to Remove Boric Acid
DB-PF-03272; Post Maintenance Valve Test; Revision 02
DB-PF-09301; Operation of Motor Monitoring Equipment; Revision 02
Pump and Valve Basis Document; Volume III; Revision 22
CR 04-05200; SW1367 Failed to Operate in Manual Mode

1R22 Surveillance Testing

DB-CH-01816: Gross Specific Activity; Revision 04

EP6 Drill Evaluation

Davis-Besse Nuclear Power Station Emergency Preparedness Integrated Drill Manual for the Drill Conducted on September 16, 2004

4OA1 Performance Indicator Verification

DB-SC-03357; RCS Water Inventory Balance; Revision 06
RCS Integrated Leakage Program Manual; Revision 01
DB-OP-01200; Reactor Coolant System Leakage Management; Revision 05
DB-PF-03010; RCS Leakage Test; Revision 05
NG-EN-00327; RCS Integrated Leakage Program; Revision 00
DB-CH-01815; Dose Equivalent I-131 Determination; Revision 00
DB-CH-01816; Gross Specific Activity; Revision 04
NOP-CC-4003; Fuel Reliability Monitoring and Assessment; Revision 01
NG-RA-00810; Reactor Oversite of NRC Performance Indicator Program

4OA2 Identification and Resolution of Problems

CR 04-000181; Missed TS Action Statement
CR 04-01230; Missed Tech Spec Entry
CR 04-02767; AF6451 Stroked Outside the Expected Range But Not Retested or Made Inoperable
CR 04-03800; DB-SC-030001 Missed Tech Spec Late Date
CR 04-04406; DP-OP-03006 Missed Surveillance Requirement 4.3.1.1.1 for RCS Flow Channel Check
CR 04-04425; Collective Significance of Tech Spec Events
NG-DB-00225; Procedure Use and Adherence; Revision 02
NOP-LP-2001; Condition Report Process; Revision 06
NOB-LP-LP-2007; Condition Report Process Effectiveness Review; Revision 01

4OA3 Event Followup

CR 03-04879; Review of Previous NRC Submittal Identifies Inaccurate Statement in LER 99-003
CR 03-05573; C&A Review of LER 99-003 Identifies Potentially Inaccurate Information
FENOC Letter 1-1324 to NRC; Notification of Information Provided to the NRC that May Not be Complete and Accurate in All Material Respects; July 15, 2003
FENOC Letter 1-1330 to NRC; Final Report: Results of Extent of Condition Review, NRC IMC 0350 Restart Checklist Item 3.i, “Process for Ensuring Completeness and Accuracy of Required Records and Submittals to the NRC”; October 24, 2003
USAR Section 3.6.2.7.1.6; Main Feedwater System
CR 03-08917; SFRCS Channels Re-energizing in a Blocked Condition

4OA5 Other Activities

CR 04-05354; Collective Significance Review of Issues Affecting Operations
CR 04-05517; Aux Transformer (X11) Current Transformer is Open Circuited
SAP Order 2000112188; DB-X11_EH: Clean/Inspect/Correct Wiring
SAP Order 200110834; PS/DM ICS Transients
CR 04-05603; EDG 1 Motor Operated Rheostat for Voltage Regulator Failure
CR 04-03639; ARTS Power Supply Trip During DB-MI-03355

LIST OF ACRONYMS USED

ADAMS	Agency-wide Document Access and Management System
AFP	Auxiliary Feedwater Pump
ARTS	Anticipatory Reactor Trip System
CFR	Code of Federal Regulations
CR	Condition Report
CSS	Containment Spray System
ECCS	Emergency Core Cooling System
FENOC	FirstEnergy Nuclear Operating Company
ICS	Integrated Control System
HPR	High Pressure Recirculation
HPI	High Pressure Injection
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IR	Inspection Report
LER	Licensee Event Report
LOOP	Loss of Offsite Power
MU	Makeup
NCV	Non-Cited Violation
NRC	United States Nuclear Regulatory Commission
PARS	Publicly Available Records
PI	Performance Indicator
RCS	Reactor Coolant System
RFO	Refueling Outage
SDP	Significance Determination Process
SFRCS	Steam Feed Rupture Control System
TS	Technical Specifications
USAR	Updated Safety Analysis Report
WO	Work Order