February 1, 2008

EA-03-214
EA-04-224
EA-07-199

Mr. Mark B. Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2, Mail Stop A-DB-3080
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2007005

Dear Mr. Bezilla:

On December 31, 2007, 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 January 8, 2008, with Mr. Kaminskas and other members of your staff. Additionally, this inspection report documents special inspection activities associated with your compliance with the Confirmatory Order EA 03-214, Confirmatory Order EA 04-224, and Confirmatory Order EA 07-199.

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.

The report documents four findings, two NRC-identified findings and two self-revealing findings, of very low safety significance (Green). Three of the findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or 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 a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Davis-Besse Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Bruce L. Burgess, Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000346/2007005
w/Attachment: Supplemental Information

cc w/encl: The Honorable Dennis Kucinich
J. Hagan, President and Chief
Nuclear Officer - FENOC
J. Lash, Senior Vice President of
Operations and Chief Operating Officer - FENOC
Manager - Site Regulatory Compliance - FENOC
D. Pace, Senior Vice President of
Fleet Engineering - FENOC
J. Rinckel, Vice President, Fleet Oversight - FENOC
D. Jenkins, Attorney, FirstEnergy Corp.
Director, Fleet Regulatory Affairs - FENOC
Manager - Fleet Licensing - FENOC
Ohio State Liaison Officer
R. Owen, Administrator, Ohio Department of Health
Public Utilities Commission of Ohio
President, Lucas County Board of Commissioners
President, Ottawa County Board of Commissioners

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Letter to M. Bezilla from B. Burgess dated February 1, 2008

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC INTEGRATED INSPECTION REPORT 05000346/2007005

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2007005
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	October 1, 2007, through December 31, 2007
Inspectors:	J. Rutkowski, Senior Resident Inspector R. Smith, Resident Inspector R. Arrighi, Senior Enforcement Specialist K. Barclay, Reactor Engineer M. Bielby, Senior Operations Engineer T. Go, Health Physicist R. Jickling, Senior Emergency Preparedness Analyst J. Neurauter, Senior Reactor Inspector G. Wright, Project Engineer
Observer:	P. Voss, Reactor Engineer
Approved by:	B. Burgess, Chief Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000346/2007005; 10/01/07 – 12/31/07; Davis-Besse Nuclear Power Station; Operability Evaluations, Refueling and Other Outage Activities

This report covers a three-month period of inspection by resident inspectors and announced baseline and supplemental inspections by regional inspectors. Four Green findings, three of which were non-cited violations (NCVs), 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 4, dated December 2006.

NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings.” Specifically, the licensee failed to provide a procedure to perform visual inspection of the polar crane structural members required by American National Standards Institute (ANSI) B30.2-1976. The issue was entered into the licensee’s corrective action program, and a licensee procedure was revised to perform visual inspection of the polar crane structural members required by ANSI B30.2-1976.

  This finding was more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the polar crane structural inspections is to limit the likelihood of a polar crane structural component failure to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems. The finding was of very low Safety significance based a Phase 1 screening in accordance with Inspection Manual Chapter (IMC) Appendix G, “Shutdown Operations Significance Determination Process (SDP),” Table 1 qualitative assessment, because no structural concerns were identified when the polar crane was inspected in the previous two refueling outages (12RFO and 13RFO) and the low number of lifts performed by the polar crane during a single refueling outage. The finding has a cross-cutting aspect in the area of human performance because the licensee did not provide a complete, accurate, and up-to-date procedure to plant personnel (H.2(c)). (Section 1R20.2)

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the licensee failed to assure and verify that the design of Internals Handling Adapter lifting pins was based on material fracture toughness as required by ANSI N14.6-1978. The issue was entered into the licensee’s corrective action program, and the licensee has initiated an engineering change to replace the Internals Handling Adapter lifting pins prior to removing the reactor vessel head in the next refueling outage 15RFO.

This finding was more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the Internals Handling Adapter meeting the design requirements of ANSI N14.6-1978 is to limit the likelihood of a structural component failure to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems. The finding was of very low Safety significance based a Phase 1 screening in accordance with IMC Appendix G, "Shutdown Operations SDP," Table 1 qualitative assessment, because although the fracture toughness of the lifting pin material was not evaluated, the lifting pins did satisfy ANSI N14.6-1978 stress design factors and the lifting pins were subjected to a low number of historical reactor vessel head lifts that utilized the Internal Handling Adapter. The finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not take appropriate corrective actions to promptly correct the design bases non-conformance identified in their design calculation (P.1(d)). (Section 1R20.2)

Cornerstone: Mitigating Systems

- Green. A self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified for failing to include appropriate quantitative or qualitative acceptance criteria for assuring the proper setting of the travel stops on valve SW-36 [Component Cooling Water Heat Exchanger 1 Service Water Outlet Valve] after valve operator maintenance. This resulted in a valve opening setting that, in the event of a safety feature system actuation, would limit service water flow to less than flows analyzed in the approved flow balance calculation for flow to the component cooling water heat exchanger 1. The licensee entered the deficiency into their corrective action program and adjusted the travel stops to provide for the proper service water flow.

  This finding is greater than minor because the finding was associated with the configuration control attribute of the Mitigating Systems Cornerstone and did affect the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding was more than minor, using example 1a of IMC 0612 Appendix E, September 20, 2007, in that testing determined CCW heat exchanger flows to be degraded subsequent to stop setting adjustment and declaring the heat exchanger operable. The finding was evaluated using the SDP and was determined to be a finding of very low safety significance because there was no actual loss of a safety system function. The finding was associated with the cross-cutting area of human performance in that the resources and specifically work packages were not adequate to ensure that work performed restored the component cooling water system to the analyzed condition (H.2(c)) after completion of maintenance activities. (Section 1R15)

- Green. A self-revealing finding of very low safety significance was identified for the licensee's failure to replace degraded emergency diesel generator (EDG) air start system hoses in accordance with operating experience (OE). Specifically, the licensee did not properly implement OE that recommended a 12-year lifespan for EDG air start hoses. This resulted in EDG2 failing to start during a monthly test due to an air leak in a hose leading to one of the air start motors. The OE was identified in 2001; at the time of the test failure, the leaking air hose had been installed on the EDG for more than 12 years. There was no violation of regulatory requirements. The licensee entered the issue into their corrective action program and replaced both the degraded hose and another similarly aged hose in the air start system.

The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance because it did not represent an actual loss of a safety function. The failure to replace the degraded hose is related to the cross-cutting element of problem identification and resolution, particularly the implementation of operating experience (P.2(b)) component in that the licensee did not implement and institutionalize relevant OE through changes to station processes, procedures, and equipment. (Section 1R15)

Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, the plant was operating at 100 percent power.

On December 15, 2007, the licensee reduced power to 95 percent to facilitate setpoint testing of main steam safety valves. Upon completion of the testing power was returned to 100 percent on December 16, 2007.

On December 30, 2007, the licensee commenced its fifteenth refueling outage. At the end of the inspection period the plant was in mode 5 with preparations ongoing to drain the reactor coolant system to the level of the reactor vessel flange.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- Service Water System; and
- Ultimate Heat Sink with emphasis on the ability to provide makeup water from Lake Erie.

This inspection constitutes one winter seasonal readiness preparations sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Decay heat train 2 on October 30, 2007, during scheduled decay heat train 1 maintenance; and
- Emergency diesel generator 2 on November 27, 2007, during a planned emergency diesel generator 1 and associated equipment outage.

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, Administrative TS, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the attachment.

These activities constituted two partial system walkdown samples as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

The week of November 15, 2007, the inspectors performed a complete system alignment inspection of the Component Cooling Water (CCW) to verify the functional capability of the system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment.

The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of past and outstanding work orders (WOs) was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program (CAP) database to ensure that system equipment alignment problems were being identified and appropriately resolved. The documents used for the walkdown and issue review are listed in the attached List of Documents Reviewed.

These activities constituted one complete system walkdown sample as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Battery rooms A and B (Fire Zone X and Y, Rooms 428A and 429B);
- Emergency diesel generator 1 room (Fire Zone K, Room 318);
- Non-radiological ventilation supply equipment room (Fire Zone II, Room 516);
- Control Room Area (Fire Zone FF, Room 502, 503, 504, 505, 506, 508, 509, 510, 511, 512); and
- Low voltage switchgear room (Fire Zone Y, Room 429, 429A, and 429B).

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee’s fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant’s ability to respond to a security event. Using the documents listed in the attachment, the 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 fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

These activities constituted five quarterly fire protection inspection samples as defined by Inspection Procedure 71111.05-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On October 17 and 19, 2007, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constitutes one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

.2 Operating Test Results (71111.11B)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the comprehensive annual job performance measure operating tests and the annual simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee during the biennial licensed operator requalification program examinations conducted in November and December 2007. The overall results were compared with the significance determination process in accordance with NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP).”

This inspection constitutes the completion of one biennial licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated the performance of the following risk significant systems:

- Reactor Coolant System/Reactor Coolant Leakage Monitoring Program; and
- Boric Acid Addition System

The inspectors reviewed events associated with the systems listed above and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed system performance with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

This inspection constitutes two quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Initial and revised risk summaries for the week of October 29, 2007, including a planned outage of decay heat train 1 and an unplanned entry into an orange risk condition due to an equipment issue that developed during testing of an auxiliary feedwater pump; and
- Initial and revised work summaries for the week on November 12, 2007, including planned outage of train 2 emergency core cooling system components.

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's work scheduling personnel, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These activities constituted two samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- CR 07-30241 - emergency diesel generator 2 failure to start due to leak in one of the air start motor supply lines;
- Past operability of service water train 1 as documented in CR 07-25993 and calculation NSA-060-05-013, “Past Operability Analysis of SW36 Mispositioning”;
- CR 07-28171 – potentially unqualified-for-harsh-environment electrical terminations in a containment motor-operated isolation valve;

- CR 06-7224 – acceptability of ultra-low-sulfur fuel oil for use in the emergency diesel generators; and
- CR 07-30534 – operability of post-accident monitoring instrument for measuring the temperature of fluid in the reactor coolant system hot leg loop 2

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and Updated Safety Analysis Report (USAR) to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the attachment.

This inspection constitutes five samples as defined in Inspection Procedure 71111.15-05.

b. Findings

(1) Component Cooling Water Heat Exchanger Service Water Flow

Introduction: A Green self-revealing NCV was identified for the improper setting of the travel stops on valve SW-36 [Component Cooling Water Heat Exchanger 1 Service Water Outlet Valve]. The setting, in the event of a safety feature system actuation, would limit service water flow to less than flows analyzed in the approved flow balance calculation.

Description: On August 31, 2007, the licensee was conducting flow balance testing of service water train 1 components. Because of the design of the plant systems, which incorporates a service water pump, a component cooling water pump and heat exchanger, and a containment air cooler that can be aligned to either of the two TS required service water trains, multiple flow tests were required to verify all potential combinations of equipment. The procedure was a new procedure that had been developed to allow online flow balancing instead of the previous norm of conducting the flow balancing during outages. The licensee had successfully completed online flow balancing of service water train 2.

The initial testing of the service water train 1 components had component cooling water (CCW) heat exchanger 1 aligned for train 1 testing. When the testing commenced, the personnel conducting the testing observed that the flow through the heat exchanger was approximately 1200 to 1500 gallons per minute (gpm) lower than they expected. The test was suspended and component cooling water train 1 was declared inoperable. The licensee entered the action statements for CCW and components cooled by CCW train 1.

The licensee, after declaring component cooling water train 1 inoperable, removed CCW heat exchanger 1 from service and aligned the swing heat exchanger, heat exchanger 3, as the heat exchanger for train 1. After observing expected service water flows through heat exchanger 3, aligned to train 1, the licensee declared CCW train 1 operable. The licensee determined that the cause of the low flows was caused by improper setting of the open mechanical stops on SW36 which is the valve on the service water outlet from CCW heat exchanger 1 and which is used to throttle flow through the heat exchanger.

SW36 is a 20 inch manual butterfly valve with a Limitorque manual operator. The operator contained mechanical stop limit devices consisting of nuts that ride on a stem. The position of the nuts can be adjusted to provide both open and close stops. The position of the open stop is determined by required periodic flow balance testing that sets the service water system such that adequate cooling water flow is delivered to system components. The last flow balancing of the service water train 1 was in April 2006 during the unit's last refueling outage. The opening lock nut was positioned to set the required valve opening to approximately 40 to 45 percent open.

In August 2007, SW36 was found to have an open limit setting less than that determined necessary by the previous flow balancing. The valve should have been able to be opened an additional three turns; approximately 50 turns are required to move the valve from full close to full open. When the valve limit stop was adjusted to permit an additional three turns, flow testing on September 4, 2007, demonstrated expected flow. After adjustment of the open stop and analysis of the data, CCW heat exchanger 1 was declared as operable on September 15, 2007.

The licensee's investigation determined that personnel replaced degraded actuator stop nuts for valve SW36 in August 2006 after the flow balancing in April 2006. Licensee work orders used to do work on the valve indicated that the replaced stop nuts were adjusted to positions consistent with those established in the April 2006 flow testing. No flow testing or flow verifications were conducted after stop nut replacement to verify that the stop nuts were properly set. The licensee concluded that the stop nuts were potentially improperly set since August 2006.

The improper setting of the SW36 stop nuts would have resulted in less than desired flow through CCW heat exchanger 1 in the event of accidents such as a loss of cooling accident during times that CCW heat exchanger 1 was aligned for train 1 service. However, the reduced flow would have provided some cooling and the throttling effect of SW36 would have caused increased flow to components cooled by service water that were in service water branch lines parallel to the CCW heat exchanger, e.g., the safety train 1 containment air cooler. At the conclusion of the inspection period, the licensee had completed an evaluation of the capability of CCW heat exchanger 1 to perform its design function. The analysis concluded that the redistribution of the service water flow was such that all post-accident design requirements would have been met for analyzed accident scenarios.

<u>Analysis:</u> The inspectors determined that the improper setting of SW36 open stop was a performance deficiency and a finding. This finding was considered more than minor because the finding was associated with the configuration control attribute of the mitigating systems cornerstone and did affect the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding was more than minor, using example 1a of IMC 0612 Appendix E, September 20, 2007, in that testing determined CCW heat exchanger flows to be degraded subsequent to stop setting

adjustment and declaring the heat exchanger operable. The finding was evaluated using the SDP and was determined to be a finding of very low safety significance because there was no actual loss of a safety system function. The finding was associated with the cross-cutting area of human performance in that the resources and specifically work packages were not adequate to ensure that work performed restored the component cooling water system to the analyzed condition (H.2(c)) after completion of maintenance activities.

Enforcement: 10 CFR 50, Appendix B, Criterion V, required that activities affecting quality shall be described by instructions or procedures that include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this, in August 2006, the licensee failed to specify and implement steps to ensure that flow through CCW heat exchanger 1 was consistent with approved flow calculations after replacement of the valve SW36 stop nuts. Because this failure is of very low safety significance and has been entered into the licensee's corrective action program as CR 07-25993, this violation (NCV 05000346/2007005-01) is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy.

(2) Emergency Diesel Generator 2 Failure to Start

Introduction: The inspectors identified a self-revealing Green finding of very low safety significance for a failure to replace degraded air start hoses for Emergency Diesel Generator 2 (EDG2) as recommended by industry operating experience (OE).

Description: On November 15, 2007, EDG2 failed to start on the DA31 air-start side (side 2) during a monthly surveillance test. EDG2 was immediately declared inoperable while operators worked to diagnose the problem. When an apparent cause was discovered, the operators attempted to start side 1 (the DA45 air-start side), which experienced a single air abutment, but subsequently started. EDG2 remained unavailable for 1.15 hours during the event. Testing on the DA31 side revealed a significant leak on an air hose feeding the pinion gear for the lower air start motor. On November 23, 2007, maintenance replaced the DA31 upper and lower air motor pinion hoses and found the hose internal rubber hard and brittle. Following testing, EDG2 DA31 air-start side was declared operable.

During investigation into the operability determination, the inspectors reviewed condition reports, corrective actions, preventative maintenance (PM) activities, TS requirements, procedures, and interviewed licensee operations and systems engineers. The inspectors found that there was operating experience that the licensee had received, that indicated that EDG air hoses should be replaced on 12-year intervals and that visual inspections of the exterior of the hoses were not sufficient to determine hose condition. The inspectors reviewed the OE used by the licensee and examined their efforts to implement it. From this review, the inspectors determined that a major contributing cause of the EDG2 side 2 air-start system failure was a licensee failure to adequately implement operating experience by not appropriately incorporating the operating experience related to EDG air-start hoses into its preventive maintenance program.

Corrective actions for July 2001 failure of the air start system on the DA30 side of EDG1 (CR 01-1795), recommended implementation of an air hose 12-year replacement preventive maintenance activity. In May 2002, the licensee revised its procedures to replace air hoses at 12-year intervals for EDG1. As of November 15, 2007, a similar revision for the EDG2 PM (DB-REV-02-0354) was still awaiting approval. The licensee had not applied the 12-year rule to the EDG2 hoses in place at the time of the 2002 PM revision request. Some of the EDG2 air hoses had been in place since the 1980's or earlier and had neither replaced nor evaluated to assess their condition. Instead, the PM revision that had been pending approval since 2002, had replacement scheduled for the next 12-year PM window in February 2013.

Analysis: Failure to replace EDG2 air start hoses after 12 years of service, as recommended, led to a significant leak due to aging on the lower air motor's hose feeding the pinion gear. This created a condition adverse to quality and resulted in EDG2 side 2 failing to start due to this leak. The hose's degraded condition caused EDG2 to be unavailable for 1.15 hours while operators determine the cause of the air start failure. Additionally, the event impacted the licensee's EDG Condition Monitoring Criteria since the EDG experienced a start failure. This failure, combined with an abutment event that occurred when the redundant air start system was tested to restore EDG operability, resulted in the licensee reaching the EDG Maintenance Rule Condition Monitoring Criteria limit of 2 per cycle.

The finding was evaluated using the SDP, and the inspectors determined that it was greater than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance because the inspectors determined that the finding did not represent an actual loss of a safety function. The failure to replace the degraded hose is related to the cross-cutting element of problem identification and resolution in that the licensee failed to implement and institutionalize EDG OE through changes to station processes, procedures, and equipment (P.2(b)). Additionally, the procedure intended to implement the air hose preventative maintenance activities (originated May 2002) had not yet been fully approved at the time of the EDG2 failure to start.

Enforcement: The inspectors concluded that the licensee failed to replace age-degraded EDG2 air start hoses in accordance with industry operating experience. These actions caused a decrease in the reliability, availability, and capability of this safety-related mitigating system and, while not representing an actual loss of safety function, led to a green finding (FIN 05000346/2007005-02). There were no violations of regulatory requirements identified. The issue was entered into the licensee's corrective action program as CR 07-30241.

1R19 Post Maintenance Testing (71111.19)

.1 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance activities for review to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Testing of make-up pump 2 on October 16, 2007, after scheduled maintenance and preventive maintenance on the motor-to-pump coupling and replacement of a time-delay agastat relay in the motor supply breaker; and
- Filling and venting decay heat train 1 and flow testing of decay heat pump 1 on November 1, 2007, after work on train 1 components that required draining a portion of the system.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion), and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the attachment.

This inspection constitutes two samples as defined in Inspection Procedure 71111.19.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Pre-Outage Activities

a. Inspection Scope

The inspectors reviewed selected portions of the Outage Plan (OP) for the licensee's fifteenth refueling outage (RFO15) and several activities necessary for the outage but that were conducted prior to the outage start date of December 30, 2007. The activities included:

- Licensee inspection of new fuel and configuration of new and stored spent fuel in the spent fuel pit to accommodate scheduled outage fuel movement;
- Licensee meeting on the potential for greater than anticipated fuel guide-tube growth and the contingencies to address if needed;
- Licensee meeting to discuss modification of decay heat system valves; and
- Licensee identification and resolution of problems related to refueling outage activities.

These inspection activities do not fully constitute one refueling outage sample as defined in Inspection Procedure 71111.20-05. The remaining activities specified in Inspection

Procedure 71111.20-05 were scheduled to be accomplished during the first quarter of 2008.

b. Findings

No findings of significance were identified.

.2 Refueling Outage Activities – Crane and Heavy Lift Inspection (OpESS FY2007-03)

a. Inspection Scope

From July 9 through November 7, 2007, the inspectors reviewed the licensee's control of heavy loads program in conjunction with the NRC's Operating Experience Smart Sample (OpESS) FY2007-03, Revision 0, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20," specifically related to the removal and installation of the reactor vessel head during refueling outages. The inspectors performed the following activities listed below during the inspection. Documents reviewed during the inspection are listed in the attachment.

- Reviewed the licensee's polar crane preventative maintenance program procedures and the polar crane manufacturer's recommended maintenance. Also reviewed a sample of licensee records of polar crane testing and inspections completed prior to reactor disassembly and reactor head lift;
- Reviewed licensee's submittals and commitments related to Generic Letters (GL) 80-113 and 81-07, "Control of Heavy Loads";
- Reviewed licensee's calculations related to a postulated reactor vessel head drop. Reviewed licensee's procedures that remove and install the reactor vessel head during refueling operations with respect to conformance to limiting parameters evaluated in the reactor head drop analysis, i.e., load drop weight, load drop height, and medium through which load drop occurs (air);
- Reviewed licensee procedures that control the total weight lifted by the polar crane to remove and install the reactor vessel head during refueling operations and the polar crane rated lift capacity;
- Reviewed licensee calculations of rigging and special lifting devices used to remove and install the reactor vessel head during refueling operations; and
- Reviewed licensee's procedures that control reactor vessel safe load path to remove and install the reactor vessel head during refueling operations.

From December 10 through December 17, 2007, the inspectors performed the following inspection activities in conjunction with the NRC's Operating Experience Smart Sample (OpESS) FY2007-03, Revision 1, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20," specifically related to the reactor vessel head removal and installation during refueling outage 15RFO:

- Reviewed licensee's revised reactor vessel head drop evaluation, Calculation C-CSS-062.01-025, "Reactor Vessel Head Drop Analysis." Reviewed licensee's procedures that remove and install the reactor vessel head during refueling operations with respect to conformance to limiting parameters evaluated in the reactor head drop analysis, i.e., load drop weight, load drop height, and medium through which load drop occurs (air); and

- Reviewed licensee's documentation associated with modifications to the reactor vessel closure head fixed lifting pendant and internals handling adapter including the engineering change package, design requirements, design calculations, manufacturing specifications, material test reports, and load test reports.

This inspection constitutes a partial completion of one refueling outage sample as defined in Inspection Procedure 71111.20 which will be completed during the next inspection interval.

b. Findings

(1) Inspection Procedure for Polar Crane Omitted Visual Inspection of Structural Components

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," having very low safety significance (Green), in that, maintenance procedures did not include inspection of polar crane structural components specified in American National Standards Institute (ANSI) B30.2-1976 standard prior to use. As a result, the licensee used the polar crane in the last refueling outage, 14RFO, without performing visual inspection of the polar crane structure.

Description: The inspectors reviewed the licensee's submittals and commitments related to Generic Letters (GL) 80-113 and 81-07, "Control of Heavy Loads." Section 9.1.5.f of the Updated Final Safety Analysis Report (UFSAR) indicates, in-part, inspecting, testing, and maintaining cranes with ANSI B30.2-1976 to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems.

The inspectors noted that polar crane procedure PM-0830 did not include ANSI B30.2-1976 requirement to inspect the structural components for deformed, cracked or corroded members, loose bolts or rivets. The licensee could not produce documentation to verify these structural inspections of the polar crane were performed during 14RFO.

In response to this concern, the licensee initiated CR 07-23369 on July 12, 2007. The licensee subsequently revised PM 0830 to include the structural inspection requirements of ANSI B30.2-1976, i.e., inspect all polar crane structural members for deformities, cracks, corrosion, and loose bolts.

Analysis: The failure to have a procedure to inspect the polar crane structural components was a performance deficiency because the licensee used the polar crane to lift the reactor vessel head over the reactor core without performing a visual inspection of the polar crane structural components during 14RFO.

The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612, Appendix B, "Issue and Screening," Minor Question 4 because the finding was associated with the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the polar crane structural inspections is to limit the likelihood of a polar crane structural component failure to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems. The inspectors with assistance from a Region III Senior Reactor Analyst (SRA) evaluated the finding using IMC 0609, Appendix G, “Shutdown Operations SDP,” Phase 1 screening. The Region III SRA determined that polar crane structural component reliability was not suitable for Significance Determination Process (SDP) analysis and performed a qualitative assessment using Appendix G, Table 1 of IMC 0609. Because no structural concerns were identified when the polar crane was inspected in12RFO and 13RFO and the low number of lifts performed by the polar crane during a single refueling outage, the SRA determined the finding to be of very low safety significance (Green). The finding has a cross-cutting aspect in the area of human performance because the licensee did not provide a complete, accurate, and up-to-date procedure to plant personnel H.2(c).

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings” requires, in-part, that activities affecting quality shall be prescribed by instructions, procedures or drawings and shall be accomplished in accordance with these instructions, procedures or drawings.

Contrary to the above, from October 22, 2004, to August 9, 2007, the licensee did not have a procedure in-place to ensure the ANSI B30.2-1976 requirement to inspect polar crane structural components was performed. Specifically, this requirement was not included in PM 0830 for the polar crane. However, because this violation was of very low safety significance and because the issue was entered into the licensee’s corrective action program (CR 07-23369), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2007005-03)

(2) Internals Handling Adapter Design Calculation Did Not Consider Material Fracture Toughness Requirements

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green), in that, the design bases analysis for the Internals Handling Adapter did not adequately evaluate the lifting pins structural component. Specifically, the calculation failed to consider the lifting pins material fracture toughness. As a result, this design basis calculation was not in conformance with design bases standard ANSI N14.6 to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems.

Description: The inspectors reviewed calculation C-CSS-062.01-024, “Internals Handling Adapter Analysis,” and noted that the calculation indicated that the design of the lifting pins should be based on its material fracture toughness properties in accordance with ANSI N14.6-1978. However, this calculation based the design of the lifting pins using stress design factors from Paragraph 3.2.1.1 of ANSI N14.6-1978 because the lifting pins were fabricated using a material of unknown fracture toughness properties. Although the licensee’s calculations identified the material fracture toughness design requirements, the licensee took no action to ensure compliance with their design bases standard, ANSI N14.6.

The inspectors reviewed the licensee’s submittals and commitments related to GL 80-113 and GL 81-07, “Control of Heavy Loads.” Section 9.1.5 of the UFSAR indicated, in-part, that details of Davis-Besse Nuclear Power Plant compliance to NUREG-0612 Phase 1 are discussed in Serial Letter 774 dated February 1, 1982.

The inspectors noted that Serial Letter 774 stipulated the Internals Handling Adapter to be in compliance with ANSI N14.6-1978, Section 3.2, “Design Criteria.” No exceptions to ANSI N14.6-1978, Section 3.2 were indicated in Serial Letter 774. Paragraph 3.2.1.1 of ANSI N14.6-1978 established stress design factors except when materials that have yield strengths above 80 percent of their ultimate strength are used. Paragraph 3.2.1.1 of ANSI N14.6-1978 further stipulated that for these materials the stress design factors do not apply, and the design shall be on the basis of the material’s fracture toughness.

Since calculation C-CSS-062.01-024 determined the existing lifting pin material yield strength to be 93 percent of the ultimate strength, an evaluation of the material fracture toughness was required to be in compliance with Section 3.2 of ANSI N14.6-1978.

In response to this concern, the licensee initiated CR 07-24954 on August 10, 2007, and CR 07-27630 on October 1, 2007. The licensee further initiated the replacement of the Internals Handling Adapter lifting pins with a material of known fracture toughness properties prior to removing the reactor vessel head in refueling outage 15RFO as part of engineering change ECP 06-0128, “Reactor Vessel Head Solid Lifting Pendant.”

<u>Analysis</u>: The inspectors determined that the failure to evaluate the material fracture toughness properties of the lifting pins was a performance deficiency because the Internals Handling Adapter was not in conformance with design bases standard ANSI N14.6-1978 design requirements.

The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612, Appendix E, “Examples of Minor Issues,” Example 3a. This issue was more than minor because in order to restore Internals Handling Adapter compliance with design bases standard ANSI N14.6-1978, a modification to the original lifting pin material was necessary. The finding was associated with the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, Compliance with ANSI N14.6-1978 is to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems. The inspectors with assistance from a Region III Senior Reactor Analyst (SRA) evaluated the finding using IMC 0609, Appendix G, “Shutdown Operations SDP,” Phase 1 screening. The Region III SRA determined that Internals Handling Adapter structural component reliability was not suitable for Significance Determination Process (SDP) analysis and performed a qualitative assessment using Appendix G, Table 1 of IMC 0609. Although the fracture toughness of the lifting pin material was not evaluated, the lifting pins did satisfy ANSI N14.6-1978 stress design factors and the lifting pins were subjected to a low number of historical reactor vessel head lifts that utilized the Internal Handling Adapter. Therefore, the SRA determined the finding to be of very low safety significance (Green). The finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee did not take appropriate corrective actions to promptly correct the design bases non-conformance identified in their design calculation P.1(d).

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” requires, in-part, that measures be established to assure that applicable regulatory requirements and the design bases, as defined in Section 50.2, are correctly translated into procedures and instructions. Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference

bounds for design. These values may be requirements derived from analysis (based on calculations or experiments) of a postulated accident for which a structure, system, or component must meet its functional goals.

Contrary to the above, on January 31, 2007, the licensee had not established effective measures to ensure that the design bases of the Internals Handling Adapter related to material fracture toughness was correctly translated into procedures and instructions. Specifically, design basis calculation C-CSS-062.01-024 did not base the design of the lifting pins on material fracture toughness as required by ANSI N14.6-1978. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (CR 07-22954 and CR 07-27630), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000346/2007005-04)

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- DB-PF-3001, Main Steam Safety Valve Setpoint Test, on December 15, 2007;
- Channel Functional Test of the Reactor Trip Breaker B, RPS Channel 1 Reactor Trip Module Logic, and ARTS Channel 1 Output Logic on October 24, 2007; and
- Station Blackout Diesel Monthly Test on October 10, 2007.

The inspectors observed in plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the USAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the attachment.

This inspection constitutes three routine surveillance testing sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.2 In-service Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Containment Spray Train 1 quarterly Pump and Valve Test on November 20, 2007.

The inspectors observed in plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the USAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers Code, and reference values were consistent with the system design basis; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the attachment.

This inspection constitutes one inservice inspection sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed a screening review of Revision 25 of the Davis-Besse Nuclear Power Station Emergency Plan to determine whether changes identified in Revision 25 decreased the effectiveness of the licensee's emergency planning for the Davis-Besse Station. This review did not constitute an approval of the changes, and as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous And Liquid Effluent Treatment And Monitoring Systems (71122.01)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the most current Radiological Effluent Release Report to verify that the program was implemented as described in Radioactive Effluent Technical Specification/Offsite Dose Calculation Manual (RETS/ODCM) and to determine if ODCM changes were made in accordance with Regulatory Guide 1.109 and NUREG-0133. The inspectors determined if the modifications made to radioactive waste system design and operation changed the dose consequence to the public. The inspectors assessed whether technical and/or 10 CFR 50.59 reviews were performed when required and whether radioactive liquid and gaseous effluent radiation monitor setpoint calculation methodology changed since completion of the modifications. The inspectors evaluated if anomalous results reported in the current Radiological Effluent Release Report were adequately resolved.

The inspectors reviewed RETS/ODCM to identify the effluent radiation monitoring systems and its flow measurement devices, effluent radiological occurrence performance indicator incidents in preparation for onsite follow-up, and the Final Safety Analysis Report (FSAR) description of all radioactive waste systems.

This inspection constitutes one sample as defined by Inspection Procedure 71122.01.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the Updated Safety Analysis Report (USAR) for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

This inspection constitutes one sample as defined by Inspection Procedure 71122.02.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walk-downs

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radwaste processing systems to verify that the systems agreed with the descriptions in the USAR and the Process Control Program and to assess the material condition and operability of the systems. The inspectors reviewed the status of radioactive waste processing equipment that was not operational and/or was abandoned in place. The inspectors reviewed the licensee's administrative and physical controls to ensure that the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed changes to the waste processing system to verify the changes were reviewed and documented in accordance with 10 CFR 50.59 and to assess the impact of the changes on radiation dose to members of the public. The inspectors reviewed the current processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized. The inspectors also reviewed the methodologies for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification in 10 CFR 61.55.

This inspection constitutes one sample as defined by Inspection Procedure 71122.02.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams, including dry active waste (DAW), spent resins and filters. The inspectors also reviewed the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The reviews were conducted to verify that the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure that the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

This inspection constitutes one sample as defined by Inspection Procedure 71122.02.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation

a. Inspection Scope

The inspectors reviewed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness. The inspectors verified that the requirements of any applicable transport cask Certificate of Compliance were met and verified that the receiving licensee was authorized to receive the shipment packages. The inspectors verified that the licensee's procedures for cask loading and closure were consistent with the vendor's approved procedures. The inspectors observed radiation worker practices to verify that the workers had adequate skills to accomplish each task and to determine if the shippers were knowledgeable of the shipping regulations and whether shipping personnel demonstrated adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H. The inspectors reviewed the training records provided to personnel responsible for the conduct of radioactive waste processing and radioactive shipment preparation activities. The review was conducted to verify that the licensee's training program provided training consistent with NRC and Department of Transportation (DOT) requirements.

This inspection constitutes one sample as defined by Inspection Procedure 71122.02.

b. Findings

No findings of significance were identified.

.5 Shipping Records

a. Inspection Scope

The inspectors reviewed six non-excepted package shipment manifests/documents completed in 2006/2007 to verify compliance with NRC and DOT requirements (i.e., 10 CFR Parts 20 and 71 and 49 CFR Parts 172 and 173).

This inspection constitutes one sample as defined by Inspection Procedure 71122.02.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports, audits and self assessments that addressed radioactive waste and radioactive materials shipping program deficiencies since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized and corrected. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also reviewed corrective action reports from the radioactive material and shipping programs since the previous inspection, interviewed staff and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution.;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of non-cited violations (NCVs) tracked in corrective action system(s); and
- Implementation/consideration of risk significant operational experience feedback.

This inspection constitutes one sample as defined by Inspection Procedure 71122.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the third quarter 2007 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program." This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Heat Removal System performance indicator for the period from the third quarter of 2006 through the third quarter of 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, MSPI derivation reports, and NRC Integrated Inspection reports for the period from the third quarter of 2006 through the third quarter of 2008 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one MSPI heat removal system sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Residual Heat Removal System performance for the period from the third quarter of 2006 through the third quarter of 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection reports for the period from the third quarter of 2006 through the third quarter of 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one MSPI residual heat removal system sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.4 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Cooling Water Systems performance for the period from the third quarter of 2006 through the third quarter of 2007. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection reports for the period from the third quarter of 2006 through the third quarter of 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one MSPI cooling water system sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Daily Review

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's corrective action program (CAP). This screening was accomplished by reviewing documents entered into the CAP 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 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a significant safety issue not identified by the licensee. The review was focused on repetitive equipment issues, but also considered the results of daily CAP item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The review included the six-month period from June 2007 through November 2007; the Davis-Besse Fleet Oversight Quarterly Performance Report (third quarter 2007); Site Roll-Up Integrated Performance Assessment (January 2007 through June 2007); Site Third Quarter Cognitive Trend Reports (July 2007 through September 2007); and issues documented in the licensee's system health reports, maintenance rule committee meeting minutes for 2007, and other documents prepared for the licensee's daily plant management meeting.

This review represented one semi-annual trend review sample as defined by Inspection Procedure 71152.

b. Assessment and Observations

No findings of significance were identified. The inspectors determined that the licensee's implementation of trending was adequate. The inspectors compared the licensee's process results with the results of the inspectors' daily screening and did not identify any discrepancies or potential trends that were not currently captured in the CAP or other licensee generated documents.

.3 Annual Sample: Review of Issues

a. Inspection Scope

The inspectors reviewed CR 07-25993, "Inadequate SW Flow Through CCW HX#1," and the associated evaluations by the licensee. The inspectors evaluated the completeness and accuracy of identification of the problem, the extent of condition, classification and resolution of the issue commensurate with its safety significance, the identification of the causes of the problem, and the appropriateness of the licensee's actions to address the problem. Additionally, because the licensee initially classified the issue as a significant condition adverse to quality, but then downgraded the issue to a condition adverse to quality, the inspectors reviewed the appropriateness of the downgrade and licensee compliance with corrective action program requirements.

This review represented one annual inspection sample.

b. Findings and Observations

On August 31, 2007, during performance of the Service Water Train 1 Design Flow Verification surveillance, the licensee discovered that service water flow through the component cooling water (CCW) heat exchanger (HX) appeared to be less than design flow. The licensee declared CCW HX 1 inoperable and investigated the cause. The licensee's investigation revealed that SW-36, the service water outlet valve of the CCW HX, was not opened enough to allow for desired flow. After the valve was repositioned and after design review of the new flow data, the licensee declared CCW HX 1operable. The licensee determined the cause of the mispositioned valve to be inadequate work planning that failed to specify an unambiguous set of key parameters to effectively maintain the correct throttled position. As a corrective action the licensee revised the Post Maintenance Test Manual to include flow verifications or flow balances following maintenance that affects the open travel stops. Additionally, the licensee planned on developing a case study to be presented to Maintenance, Planning, Operations and engineers that emphasizes communications of critical parameters and the specific expected results to be achieved during testing following maintenance. The case study was intended to also convey the need for a questioning attitude and the need to stop to seek assistance when expected results are different then expected.

c. Conclusions

The inspectors verified the adequacy of the following aspects of Condition Report 07-25993, associated with the inadequate SW flow through CCW HX#1: the completeness and accuracy of identification of the problem, the extent of condition, classification and resolution of the issue commensurate with its safety significance, the identification of the causes of the problem, and the identification of corrective actions. In addition, the inspectors reviewed the licensee's planned long-term corrective actions for adequacy.

No findings of significance were identified.

.4 Annual Sample: Review of Issues

a. Inspection Scope

The inspectors reviewed the licensee's response to Confirmatory Order EA 07-199 issued on August 15, 2007, and specifically reviewed the Regulatory Sensitivity Training provided to senior Davis-Besse personnel.

This review represented one annual inspection sample.

b. Findings and Observations

On October 30, 2007, the inspectors observed the training provided to senior Davis-Besse personnel and reviewed the material used in the training. The training instructor was able to present the material such that few questions were asked by the participants. When questions were asked the instructor was able to provide answers. The inspectors also observed that the training material covered the reasons for the training and copies of appropriate historical documents were included in the provided training material. That material included copies of various confirmatory orders, notice of violations, and replies to notices of violations during the period of 2004 to the present.

c. Conclusions

The inspectors concluded that the training and material presented was adequate for enhancing the understanding of Davis-Besse participants on the potential regulatory sensitivity of actions and activities undertaken by the licensee and its corporate offices.

No findings of significance were identified.

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 Decreasing Ultimate Heat Sink Level

a. Inspection Scope

On November 6, 2007, a low service water forebay annunciator alarm was received in the main control room when the water level in the forebay reached an elevation of approximately 564 feet and was decreasing. The normal water elevation is at approximately 569 feet. The service water forebay was designed as the station's ultimate heat sink. Technical Specifications require the level to be maintained at or above an elevation of 562 feet. The inspectors reviewed the licensee's response to the alarm including their use of procedures written to address low and decreasing water levels. The inspectors also reviewed that licensee's conclusion that high winds caused a decrease in lake level and that the decrease in forebay level, and subsequent return to normal level, was attributable to changes in lake level.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Contaminated Gravel By Railroad Tracks Inside the Protected Area

a. Inspection Scope

On October 22, 2007, the licensee, while conducting a standard radiological survey of railroad ties that were being replaced, found a small area of contaminated gravel next to the plant's railroad tracks. The area was inside the licensee's protected area. The inspectors reviewed the licensee's activities to quantify the volume of material and the level of activity and to determine the extent of condition. Additionally the inspectors reviewed the licensee's activities to determine the source of the contamination. The licensee did not have any record of a radioactive spill in this area but was aware the reactor vessel head that had been replaced in 2002 had been stored for a period of time in the vicinity of the location with the contaminated gravel.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 (Closed) Licensee Event Report (LER) 05000346/2006-004-02: Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures Davis-Besse Nuclear Power Station, Unit No.1

LER 05000346/2006-004 Revision 0 and Revision 1 were closed in Inspection Report 05000346/2007003.

LER 05000346/2006-004 revision 2, submitted on August 29, 2007, revised the LER and previous commitments based on evaluations performed following determination of the safety significance of the issue. This revision also updated the corrective actions to include changing the Davis-Besse Design Criteria Manual to add a statement that safety related ventilation systems and their components must be designed for applicable design basis tornado differential pressures. Additionally, the corrective actions updated the Design Interface Review Checklist to ensure that personnel performing design interface evaluations as part of engineering activities properly addressed tornado differential pressure loads that may affect safety-related ventilation systems.

LER 05000346/2006-004-02 is closed.

This review represented one inspection sample.

.4 (Closed) Unresolved Item (URI) 05000346/2007004-02: Reduced Flow Through Component Cooling Water 1 Heat Exchanger Because of Improper Valve Opening Limit Stop

This event, which occurred during the third quarter of 2007, was described in Inspection Report 05000346/2007004, and involved less than anticipated service water flow through component cooling water heat exchanger 1. The cause of the low flow was due to improper adjustment of a valve operator open-limit stop nut after a maintenance activity on the service water outlet valve to the component water heat exchanger 1. During the third quarter inspection interval the licensee and NRC had not completed a determination of safety significance of the issue. Section 1R15 of this report documented that the safety significance determination was completed and reviewed by the inspectors. That section also stated that the inspectors determined that the issue was a violation of regulatory requirements. Section 4OA2 of this report documented a review of the condition report and root cause report for the issue. This URI is closed.

4OA5 Other Activities

.1 Licensee Activities and Meetings

The inspectors observed select portions of licensee activities and meetings and met with licensee personnel to discuss various topics. The activities that were sampled included:

- Davis-Besse Weekly Outage Management Team meeting on October 17, 2007, during which there was a discussion regarding the replacement of decay heat exchangers' outlet valves (DH14A, DH14B) and bypass valve (DH13B) which would result in the closure of open operable determination;
- Increases in reactor coolant system unidentified leakrate that exceeded the action level limits of IMC 2515, Appendix D, "Plant Status" which were discussed with the licensee on December 23, 2007 and December 26, 2007; and
- Plant Review Committee meeting on December 26, 2007, to review and approve a draft licensee event report.

.2 In-Process Observation of the 2006 Safety Culture/Safety Conscious Work Environment Independent Assessment Activity

a. Inspection Scope

By letter dated July 14, 2006, FENOC addressed the NRC's March 2004 Confirmatory Order requirement for Davis-Besse to perform an annual independent assessment of safety culture/safety conscious work environment (SC/SCWE). The letter stated that the 2007 SC/SCWE assessment would be conducted by Synergy Consulting Services Corporation (Synergy).

As part of the NRC's continuing oversight inspection activities at Davis-Besse, the inspectors observed the assessment team's evaluation of information gathered during three days of one-on-one interviews. The inspectors noted that Synergy had scheduled over 100 one-on-one interviews. In addition, the inspectors observed the independent team's final briefing of the licensee on the overall results of the assessment.

In addition to observing the Synergy team, the inspectors also observed the licensee's implementation of its Business Practice, NOBP-LP-2501, Safety Culture Assessment, Revision 8. The observation was to provide input to the assessment of the licensee's self assessment activities.

b. Observations and Findings

The three-person Synergy team reviewed information gathered during the interviews, assessed how the information correlated with information from other interviews and with data obtained from a written survey. In addition, the Synergy team identified areas to address during subsequent interviews. The inspectors concluded that the Synergy team appropriately evaluated individual interview results against other interviews and information obtained through the written survey. In addition, the Synergy team used the information to focus future interviews to gain additional insights into areas of interest. The final report on the independent SC/SCWE assessment is expected to be submitted to the NRC by February 2008 and will be reviewed at that time.

The inspectors identified that the licensee had made a number of changes from Revision 7 of its Business Practice, NOBP-LP-2501, Safety Culture Assessment. The inspectors noted that the licensee continues to not apply weighting factors to individual questions in its roll-up calculations thus all questions and organizations are of equal weight regardless of the area being assessed. In addition, almost 60 percent of the questions are evaluated by numbers from surveys or meetings which did not lead to any discussion by the licensee's management team. Overall, the inspectors concluded that the licensee's SC/SCWE assessment process has not substantially improved since the NRC first reviewed it in late 2003, i.e., many of the issues noted by the NRC team inspection in 2003 remained in the current version of the Business Practice. In addition, the inspectors concluded that the process had digressed to some extent in that the group discussions, the function considered most valuable by the NRC in 2003, observed in 2003 and 2004 were not as visible in 2007.

.3 Review of the 2006 Corrective Action Program Independent Assessment Activity

a. Inspection Scope

The inspectors reviewed the licensee's independent assessment plan for the 2007 Corrective Action Program Independent Assessment. The inspectors reviewed the assessment plan and the roster of individuals that conducted the assessment contained in the licensee's June 11, 2007 letter, and the final assessment report dated September 17, 2007. In addition, the inspectors observed the independent team's activities during its assessment activities. The reviews were conducted to assess whether the independent assessment was consistent with the plan, whether the team was independent from the site and corporate headquarters, and whether areas for improvement (AFI) were appropriately addressed.

b. Observations and Findings

The 2007 Corrective Action Program Independent Assessment plan included the following areas:

- Identification, classification, and categorization of conditions adverse to quality;
- Evaluation and resolution of problems;
- Corrective action implementation and effectiveness;
- Trending program Implementation and effectiveness;
- Impact of program backlogs;
- Effectiveness of internal assessment activities;
- Open corrective actions proposed in response to the NRC Special Team Inspection - Corrective Action Program Implementation - NRC Inspection Report 05000346/2003010; and
- Corrective actions taken in response to the areas for improvement (AFI) and areas in need of attention (ANA) identified during the previous independent assessment of the Davis-Besse corrective action program implementation.

The review concluded that the scope of the plan and the individuals who were selected to perform the independent assessment were appropriate.

At the conclusion of the 2007 Corrective Action Program Independent Assessment activities, the inspectors observed the independent assessment team debriefing with the licensee concerning the assessment results. The licensee submitted the final report for the "Independent Assessment Report of the Corrective Action Program Implementation for the Davis-Besse Nuclear Power Station – Year 2007." The independent assessment team concluded that the licensee's overall implementation of the corrective action program was effective. Of the general areas assessed, seven were rated as Effective and two were rated as Highly Effective. No Areas-For-Improvement (AFI) were identified. The one AFI identified in the 2006 assessment was elevated to an area in need of attention (ANA) based on the licensee implementing its new trending program; however, the program had yet to generate a report.

The independent assessment team identified several ANAs. An ANA was defined as an identified performance, program, or process element within an area of assessment that, although sufficient to meet its basic intent, management attention was required to achieve full effectiveness and consistency. The ANAs were not required to be addressed by formal Action Plans submitted to the NRC, but were entered into the corrective action program by the licensee. For completeness, the inspectors reviewed the condition reports associated with the ANAs and identified no issues.

Based on the reviews and observations, the inspectors concluded that the 2007 independent assessment of the licensee's corrective action program was conducted by individuals independent of the licensee's organization, that the assessment team's members were all qualified to perform the assessment, that the assessment was conducted in accordance with the licensee's plan, and that issues identified by the assessment had been appropriately addressed through the corrective action program.

The inspectors did note that the independent assessment team had not performed an assessment of how well the licensee's corrective action program handled human performance issues. The licensee acknowledged the inspectors observation and indicated it would review the issue for the 2008 independent assessment.

.4 In-Process Observation of Corrective Actions Associated with the NRC's August 15, 2007 Confirmatory Order.

a. Inspection Scope

By letter dated August 15, 2007, the NRC issued an immediately effective Confirmatory Order EA-07-199 (Order) that formalized commitments made by the FirstEnergy Nuclear Operating Company (FENOC). FirstEnergy Nuclear Operating Company's commitments were documented in its July 16, 2007, letter responding to the NRC's May 14, 2007, Demand for Information (DFI).

The DFI was issued in response to information provided by FENOC relative to an analysis performed by Exponent Failure Analysis Associates and Altran Solutions Corporation into the 2002 Davis-Besse reactor pressure vessel head degradation event. On June 13, 2007, FENOC provided its response to the DFI and on June 27, 2007, the NRC held a public meeting with FENOC to discuss the DFI response. On July 16, 2007, FENOC provided a supplemental response to the DFI that provided additional detail regarding the planned implementation of commitments established in the June response to the DFI.

In addition to implementing interim corrective actions, the Order required the licensee to:

- Conduct regulatory sensitivity training for selected FENOC and non-FENOC First Energy employees to ensure those employees identified and communicate information that has the potential for regulatory impact either at FENOC sites or within the nuclear industry to the NRC. The licensee was to provide the population to be trained, the training methodology and materials, and the training objective at least 30 days prior to conducting the training. All training was to be conducted by November 30, 2007;
- Conduct effectiveness review to determine if an appropriate level of regulatory sensitivity was evident among First Energy employees including those who received regulatory sensitivity training in January 2008 and 2009;
- Develop a formal process to review technical reports prepared as part of a commercial matter. The process was to be implemented no later than December 14, 2007;
- Assess its Regulatory Communications Policy and make process changes to its NRC correspondence procedure to ensure specific questions are asked during the process relative to the experience gained from efforts to respond to the NRC's May 14, 2007, Demand for Information. Revisions were to be completed by December 14, 2007;
- Provide an Operating Experience (OE) document to the nuclear industry by September 15, 2007; and
- Complete a root cause evaluation of the events that culminated in the issuance of the May 14, 2007, DFI and provide the NRC with a summary of the analysis no later than December 14, 2007.

To assess the licensee's activities associated with item 1, i.e., conduct regulatory sensitivity training, the inspectors reviewed the licensee's training material; class hand-outs; qualifications of the individual providing the training; training objectives: and the basis for the individuals selected to receive the training. In addition, the inspectors observed the training provided to FirstEnergy and FENOC individuals on November 5 and 16, 2007, at FirstEnergy Headquarter in Akron, Ohio.

To assess the training's short-term effectiveness, the inspectors conducted interviews on December 18 and 19, 2007, with non-FENOC FirstEnergy individuals who had participated in the training.

Bulleted items 2 through 6 will be documented in future inspection reports.

b. <u>Observations and Findings</u>

Based on the documentation reviews, discussions, observations, and interviews, the inspectors concluded that for the Regulatory Sensitivity training:

> The licensee had provided the requisite material 45 days prior to the start of the training via a September 20, 2007, letter from J. Hagen, FENOC, to C. Carpenter, NRC. That letter provided, by title, the individuals, FENOC and non-FENOC FirstEnergy, who where selected to receive the sensitivity training. The letter also identified that the training would be provided in a "classroom/small group setting using lecture and case studies." In addition, eleven "enabling objections" were identified. The letter provided an outline of the training

indicating the basic areas to be covered, including: Regulations, Safety Culture, Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head degradation Event Case Study, Institute of Nuclear Power Operations Performance Objectives and Criteria, Exponent Report Case Study, Process, Hypothetical Case Studies, and a Check for Understanding.

The inspectors' review of the training material and observations of the training concluded that as a first cut, the individuals designated to receive the training were appropriate. Further, the selection was made by individuals knowledgeable of FirstEnergy processes and the causes for the DFI being issued. The training covered all the enabling objectives and generally followed the outline. The training material was appropriate and well organized to lead individuals not familiar with the nuclear industry through the reasons for the DFI and subsequent Confirmatory Order. The presenter was very knowledgeable of materials presented and presented the material at a comfortable pace.

By providing an overview of the regulatory process, concepts of safety culture and safety conscious work environment, operating experience and parallels with the handling of the Exponent Report, the training was appropriate to instill in the individuals a sensitization to issues which may impact FirstEnergy's nuclear facilities and may be of interest to the NRC and the industry.

While the presenter invited group participation during the presentation, active participation by the group was limited for the training sessions observed by the inspectors. However, when the presenter solicited feedback on areas where the training might be applied, he received a number of ideas from various individuals, indicating that the training had been effective in delivering its overall message. This was confirmed during the individual interviews conducted in mid-December, where all individuals were able to articulate the overarching message the training had been designed to deliver.

In addition to the training provided at FirstEnergy Headquarters in Akron, training was provided to selected individuals at each of FENOC's reactor sites. Those training sessions were observed by the NRC's resident staff at Davis-Besse, Perry, and Beaver Valley. Details of those observations can be found in Inspection Reports 005000346/2007005, 005000440/2007005, and 05000334 and 05000412/2007005 respectively.

.5 VERIFICATION OF ACTIONS TAKEN IN RESPONSE TO FENOC CONFIRMATORY ORDER EA-04-224

a. Scope

On July 15, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued a Confirmatory Order (EA-04-224) to the FirstEnergy Nuclear Operating Company (FENOC). The Confirmatory Order actions were agreed upon by FENOC and the NRC during an alternative dispute resolution (ADR) session held on May 11, 2004, to resolve NRC concerns regarding whether a violation of employee protection requirements occurred at the Davis-Besse Nuclear Power Station (Davis-Besse). The actions focused on providing safety conscious work environment (SCWE) training to contractor personnel who are granted unescorted access to Davis-Besse and the other FENOC nuclear facilities. In a letter dated October 4, 2005, FENOC provided the NRC with the actions the company had taken as required by the Order. An enforcement specialist from the Office of Enforcement reviewed the actions outlined in the letter to verify that they satisfied the conditions specified in the Order.

b. Observations and Finding

The specialist reviewed training module (CON-PWE-1002) that was provided to the Davis-Besse food service contractor management and supervision to verify it adequately addressed SCWE and 10 CFR 50.7, “Employee protection,” requirements; as well as the FENOC Plant Access Training module to ensue all site personnel are trained on SCWE policies. In addition, the specialist reviewed records from the FENOC Integrated Training data base to verify that the Davis-Besse food services contractor manager received the required SCWE training and reviewed records from the SCWE department to verify that contractors at Davis-Besse and the other FENOC nuclear facilities participated in the annual SCWE surveys as required by the Order.

c. Conclusion

The review concluded that: SCWE training provided an adequate overview of employee protection requirements and the elements of a good SCWE; FENOC provided the SCWE training to the Davis-Besse food services contractor manager; contractor personnel are participating in the annual SCWE audits; and the Confirmatory Order is properly being implemented.

4OA6 MANAGEMENT MEETINGS

.1 Exit Meeting Summary

On January 8, 2008, the inspectors presented the inspection results to Mr. Kaminskas and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

On November 7, 2007, the inspectors presented Polar Crane and Heavy Lift inspection results to Mr. M. Bezilla and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that licensee design calculations generated by contractors were considered proprietary. It was agreed that all paper copies of these proprietary documents would be shredded, and all electronic files of these proprietary documents would be deleted.

Additional interim exits were conducted for:

- Emergency Preparedness inspection with Mr. J. Vetter on December 18, 2007;
- Biennial Operator Requalification Program Inspection with Mr. D. Lange on December 12, 2006; and
- Radioactive Material Processing and Transportation Inspection with Mr. V. A. Kaminskas, on December 13, 2007.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Bezilla, Site Vice President
R. Bair, Staff Engineer, Mechanical/Structural
B. Boles, Director, Maintenance
K. Byrd, Manager, Design Engineering
A. Garza, ALARA Radiation Protection
S. Gatter, Liquid Radwaste System Engineer
J. Grabnar, Director, Engineering
L. Harder, Radiation Protection Manager
J. Hook, Design Engineering Supervisor
R. Hovland, Manger, Technical Services
R. Hruby, Manager, Nuclear Oversight
V. Kaminskas, Director, Plant Operation
J. Noble, Lead Radiation Protection
A. Percival, Adv. Nuclear Specialist (Chemistry)
S. Plymale, Manager, Plant Engineering
C. Price, Director, Performance Improvement
B. Reineck, Senior Engineer, Mechanical/Structural
J. Reuter, Radwaste Supervisor/Shipping
J. Rinckel, Vice-President, Fleet Oversight
J. Scott, Staff Nuclear Specialist
J. Sturdavant, Regulatory Compliance
S. Trickett, Supt., Radiation Protection
J. Vetter, Emergency Response Manager
G. Wolf, Staff Engineer, Regulatory Compliance
D. Wuokko, Acting Manager, Regulatory Affairs
K. Zellers, Supervisor, Analysis Group and Design
B. Zibung, Fleet Oversight Assessor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000346/2007005-01	NCV	Reduced Flow Through Component Cooling Water 1 Heat Exchanger Because of Improper Valve Opening Limit Stop
05000346/2007005-02	FIN	Failure to Implement Relevant Operating Experience Results in Emergency Diesel Gen
05000346/2007005.03	NCV	Inspection Procedure for Polar Crane Omitted Visual Inspection of Structural Components
05000346/2007005.04	NCV	Internals Handling Adapter Design Calculation Did Not Consider Material Fracture Toughness Requirements

Closed

05000346/2007004-02	URI	Reduced Flow Through Component Cooling Water 1 Heat Exchanger Because of Improper Valve Opening Limit Stop
05000346/2006-004-02	LER	Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures

LIST OF DOCUMENTS REVIEWED

The following is a partial 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 sections or 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 this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

Condition Reports:
- Field Observation Card DBF200x-xxxx; xxxxxxx; Xxxxx xx, 200x
- Manager-Operations Memorandum; Xxxxxx Xxxxxxx Revision x; dated Xxxxxxx xxx, 200x
- CR 07-27789; Delays in Implementation in BWST Freeze Protection Modification
- CR 02-06569; Heat Trace and Freeze Protection System Degradation and Aging
- GEO. Gradel Company contract description; contracted pumping equipment on standby

Procedures:
- NG-DB-0xxxx; Xxxxxx xxxxxxxxxxxProgram; Revision 7
- DB-OP-06913 Seasonal Plant Preparation Checklist; Revision 16
- DB-OP-06331; Freeze Protection & Electrical Heat Trace; Revision 17

Work Orders:
- WO200163366; ECR 03-0619 – install redundant ckt for 90
- WO200163367; ECR 03-0619 – install redundant ckt for 91
- WO200163353; ECR 03-0619 – install redundant ckt for 88
- WO200163363; ECR 03-0619 – install redundant ckt for 89
- WO200163372; ECR 03-0619 – install redundant ckt for 4 & 12
- WO200281874; P95 – Replace pump and motor
- WO200257508; P94 – Replace bearing assembly (pump)

1R04 Equipment Alignment

Condition Reports:
- CR 07-18312; Oil Leak on the #3 CCW Pump Outboard Motor Bearing
- CR 07-12779; MP43-1 (CCW Pump Motor #1) Surge Capacitor as Found Condition
- CR 06-6749; CC1467 CCW from Decay Heat Cooler 1 Solenoid Outlet Valve Would Not Close
- CR 07-27010; CC1495 Open Stroke Delay
- CR 07-20940; P43-2 Component Cooling Water Pump/Motor Coupling Gap
- List of Condition Reports from the last twelve months for Component Cooling Water System

Procedures:
- DB-OP-6012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 29
- DB-OP-06262; Component Cooling Water System Procedure; Revision 16

Work Orders:
- List of Open Work Orders for Component Cooling Water System on October 31, 2007

Drawings:

- Drawing OS-4, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 43
- Drawing M-036, Sheet A; Piping & Instrument Diagram Component Cooling Water; Revision 28
- Drawing M-036, Sheet B; Piping & Instrument Diagram Component Cooling Water; Revision 34
- Drawing M-036, Sheet C; Piping & Instrument Diagram Component Cooling Water; Revision 27
- Drawing OS-41A, Sheet 1; Emergency Diesel Generator Systems; Revision 27
- Drawing OS-41A, Sheet 2; Emergency Diesel Generator Systems; Revision 25
- Drawing OS-41B; Emergency Diesel Generator Air Start/Engine Air System; Revision 33
- Drawing OS-41C; Emergency Diesel Generator Diesel Oil System; Revision 16

1R05 Fire Protection

Condition Reports:
- CR 07-28655: NRC Concern About In #2 EDG Room and Suppression System Function (NRC Identified)
- CR 07-29254; Drawing A-225F Contains an Error for the Type of Extinguisher Located At F-7 (NRC Identified)

Procedures:
- DB-FP-00007; Control Of Transient Combustibles; Revision xx
- DB-FP-00009; Fire Protection and Fire Watch; Revision xx
- DB-OP-02529; Fire Procedure; Revision xx
- Davis-Besse Nuclear Power Station Fire Hazard Analysis Report

Drawings:
- Drawing A-225F; Fire Protection General Floor Plan EL 623'; Revision 14
- Drawing A-224F; Fire Protection General Floor Plan EL 603'; Revision 21
- Drawing A-223F; Fire Protection General Floor Plan EL 585'; Revision 18

1R11 Licensed Operator Requalification Program

Condition Reports:
- CR 07-28728; Crew simulator Evaluation Failure
- Simulator Guide ORQ-EPE-S111; High Seal Inj flow, RCP Shutdown Due to High Vibs, Small RCS Break with an AFW Level Control Malf; Revision 11
- Simulator Guide ORQ-EPE-S113; Transformer Oil Leak and Lockout, CRD Booster Trip, RCS Leak, All HPI Lost

Procedures:
- DBBP-TRAN-0017; Conduct of Simulator Training; Revision 03
- DBBP-TRAN-0502; Development and Conduct of Continuing Training Simulator Evaluations; Revision 04

1R12 Maintenance Effectiveness

Condition Reports:
- CR 05-02165; RCS Unidentified Leakage Rise From Approx. 0.024 to 0.26
- CR 06-00122; Breaker BPH501 Found Tripped, With Closed Indication
- CR 06-01767; RCP Motor 2-2 Stopped Rotating While Jacking For Alignment

- CR 06-02042; PORV Leaking After PROV Cycle Test DB-SP-03363
- CR 06-02192; RCP 2-1 Lower Brng Oil Level High
- CR 06-02263; Pressurizer Quench Tank Level Indicator is Behaving Erratically
- CR 06-10201; BACC Cycle 15 Mini Outage: Boric Acid Found On RC14A
- CR 06-10496; BACC; Boric Acid Deposits were Found at the Packing Area MU282
- CR 07-16429; RCP 2-2 Seal Standpipe Level Alarm (6-4-D) and Computer point (L854) Locked In
- CR 07-19039; RCP 2-1 Thrust Bearing Oil Level Low
- CR 07-22491; Leakage Trends
- CR 07-22740; Trending CR For Demin Flush Flow to RCP 1-1
- CR 07-26650; Reactor Coolant System Leakrate Anomaly

Procedures:
- DB-OP-01200; Reactor Coolant System Leakage Management; Revision 08
- DB-OP-06900; Plant Heatup; Revision 36
- DB-SP-03557; RCS Water Inventory Balance; Revision 11
- EN-DP-01171; Engineering Implementation of RCS Integrated Leakage Program; Revision 01
- NG-EN-00327; RCS Integrated Leakage Program; Revision 00

Other:
- DB System Health Report, Reactor Coolant System Window; Second Quarter, 2006
- DB System Health Report, Reactor Coolant System Window; Third Quarter, 2006
- DB System Health Report, Reactor Coolant System Window; Fourth Quarter, 2006
- DB System Health Report, Reactor Coolant System Window; First Quarter, 2007
- DB System Health Report, Reactor Coolant System Window; Second Quarter, 2007
- DB System Health Report, Reactor Coolant System Window; Third Quarter, 2007
- DB System Health Report, Reactor Coolant System Window; Second Quarter, 2007
- DB System Health Report, Reactor Coolant System Window; Third Quarter, 2007
- DB System Health Report, Boric Acid System Window; Second Quarter, 2007
- DB System Health Report, Boric Acid System Window; Third Quarter, 2007
- Maintenance Rule Expert Panel Minutes February 9, 2006
- Maintenance Rule Expert Panel Minutes May 9, 2006
- Maintenance Rule Expert Panel Minutes June 15, 2006
- Maintenance Rule Expert Panel Minutes August 10, 2006
- Maintenance Rule Expert Panel Minutes September 14, 2006
- Maintenance Rule Expert Panel Minutes October 12, 2006
- Maintenance Rule Expert Panel Minutes November 9, 2006
- Maintenance Rule Expert Panel Minutes December 14, 2006
- Maintenance Rule Expert Panel Minutes January 11 2007
- Maintenance Rule Expert Panel Minutes February 8, 2007
- Maintenance Rule Expert Panel Minutes March 8, 2007
- Maintenance Rule Expert Panel Minutes June 14, 2007
- Maintenance Rule Expert Panel Minutes July 12, 2007
- Maintenance Rule Expert Panel Minutes September 13, 2007
- Operational Decision Making Issue; If the Low Oil Level Indication for RCP 2-1 Upper Reservoir Comes into Alarm; Revision 01
- OPS-JIT-S414; JITT For Motor 1-1 Up Thrust Temperature Rise
- Problem Solving Plan for CR 06-02042; Pilot Operated Relief Valve (PORV) Leaking after PORV Cycle Test DB-SP-03363
- RCS Integrated leakage Program – Chronological Log (Cycle 15)

1R13 Maintenance Risk Assessments and Emergent Work Control

Other:
- Maintenance Risk Summaries for the Week of October 29, 2007; Revisions 0, 1, and 2
- Maintenance Risk Summaries for the Week of November 12, 2007; Revisions 0, and 1
- Operations Evolution Order; Return DH Train 1 to Service After Maintenance; October 31, 2007
- Clearance NDB-Sub049-02-025; Decay Heat Outage for Week of October 29, 2007;
- Work Implementation Schedule, Subsystem Sort; October 31, 2007
- Work Implementation Schedule, Subsystem Sort; November 13, 2007

Procedures:
- DBBP-OPS-0003; On-line Risk Management Process; Revision 6
- NOP-OP-1007; Risk Determination; Revision 5

Drawings:
- Drawing OS-4, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 43

1R15 Operability Evaluations

Condition Reports:
- CR 07-30241; EDG #2 Starting Issues
- CR 07-25993; Inadequate SW Flow Through Heat Exchanger #1
- CR 07-26744; Questionable Motor Terminations on MV5010D
- CR 07-28171; Unqualified Motor Terminations in CV5010B
- CR 06-7224; Ultra Low Sulfur Diesel Fuel Evaluation for DB
- CR 07-30534; RC Loop 2 HLG WR Temperature Indicator TIRC3A5 Exceeds TS Tolerance
- Technical Specifications 3/4.8; Electrical Power Systems; Amendment Nos. 203, 206, 219 & 273
- USAR Section 8.3; Onsite Power Systems; Revision 20
- CR 01-1795, EDG Air Start failure
- CR 07-31128, Replacement of old EDG flex hoses: CR 07-30241 Extent of condition
- List of Condition Reports for 2001-2007 for EDG air start system

Procedures:
- NOP-LP-2001, Corrective Action Program Revision 17
- NOP-LP-2100, Operating Experience Program Revision 1

Work Orders:
- Davis-Besse Nuclear Power Plant Maintenance Plan: Preventative, Plan No. 5695
- Davis-Besse Nuclear Power Plant Maintenance Plan: Preventative, Plan No. 5600

Drawings:
- Operational Schematic OS-0041A; Emergency Diesel Generator System; Revision 27
- Operational Schematic OS-0041B; Emergency Diesel Generator Air Start/Engine Air Start Systems; Revision 34

Calculations:
- Calculation C-NSA-060.05-013; Past Operability Analysis of SW036 Mispositioning; Revision 0

Other:
- Notification 600346187; CR 06-7224 ULSD Diesel Fire Pump
- Notification 600346182; CR 06-7224; ULSD Misc & ERF Diesel
- Notification 600337828; C=IN 2006-22 Ultra-Low-Sulfur Diesel Fuel
- Problem Solving Plan for CR 07-30534; December 6, 2007
- Davis-Besse Plant Health Report 3rd Quarter 2007, System 07-01 EDG
- Maintenance Rule Program Manual DB-PF-00003 EDG
- Repetitive Maintenance - Revision Request Form DB-REV-02-0353
- Repetitive Maintenance - Revision Request Form DB-REV-02-0354
- Operating Experience related to EDG air start hoses

1R19 Post Maintenance Testing

Condition Reports:
- CR 07-28665; MU PMP 1-2 Failed Agastat Relay

Procedures:
- DB-OP-3136; Decay Heat Train 1 Pump and Valve Test; Revision 15
- DB-SP-3212; Venting of ECCS Piping; Revision 9

Work Orders:
- WO200005240; DH31: Repack, Weld Leak-Off Plug, Disassemble, Replace the Bonnet Gasket and Inspect Valve

Drawings:
- Drawing OS-4, Sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 43

1R20 Outage Activities

Polar Crane and Heavy Lift Inspection (OpESS FY2007-03)

Condition Reports:
CR 02-00769; Reactor Head Lift Height Above Vessel May Violate NRC Commitments; dated February 24, 2002
- CR 02-02659; “9PNG” Statused Commitments Continue to Be of Concern; dated June 18, 2002
- CR 07-20145; Reactor Vessel Head Drop Analysis; dated May 8, 2007
- CR 07-23076; Discrepancies Found with Closed CA No. 02-00769; dated July 6, 2007
- CR 07-23106; USAR Description for Control of Heavy Loads (Reactor Head Movement); dated July 6, 2007

Condition Reports Initiated as a Result of NRC Inspection:
- CR 07-23369; Polar Crane Preventative Maintenance Requires Structural Inspection; dated July 12, 2007
- CR 07-23483; NRC Heavy Loads Inspection - 50.59 Evaluation of RV Head Drop Analysis; dated July 13, 2007
- CR 07-24954; NRC Reactor Vessel Head Drop Inspection Comments/Concerns with Calculation C-CSS-062.01-024; dated August 10, 2007

Procedures:
- DB-MM-04004; Mechanical Maintenance Procedure: Station Cranes Periodic Test; Revision 06
- DB-MM-04010; Periodic Test Procedure: Special Lifting Devices; Revision 06
- DB-MM-04010; Periodic Test Procedure: Special Lifting Devices; Revision 07
- DB-MM-06002; Mechanical Maintenance Procedure: Polar Crane Operation; Revision 11
- DB-MM-00242; Mechanical Maintenance Procedure: P&H Station Crane Maintenance; Revision 02
- DB-MN-00006; Administrative Procedure: Control of Lifting and Handling; Revision 09
- Document M-83-74-3; Harnischfeger Overhead Cranes Instruction Manual; Bulletin C-7-3
- Document 03-5016126; Procedure: Reactor Vessel Head Removal; Revision 3
- Document 03-5016141; Procedure: Reactor Vessel Head Reinstallation; Revision 2
- Document 23-9067895; Quality Assurance Package: Reactor Vessel Closure Head Fixed Lifting Pendants and Associated Cover Plates and Replacement Internals Handling Adapters; Revision 0
- Document 27-9016007; Manufacturing Specification: Reactor Vessel Head Fixed Lifting Pendants; Revision 1
- Document 27-9040002; Manufacturing Specification: Replacement Parts for DB-1 Internals Handling Adapters; Revision 1
- Document 51-9030308; Engineering Information Record: DB-1 Design Requirements for RV Head Fixed Lifting Pendants; Revision 0
- Document 51-9030308; Engineering Information Record: DB-1 Design Requirements for RV Head Fixed Lifting Pendants; Revision 2
- Document 51-9033911; Engineering Information Record: Analytical Requirements for Original DB-1 Internals Handling Adapter; Revision 1
- Document 51-9033911; Engineering Information Record: Analytical Requirements for Original DB-1 Internals Handling Adapter; Revision 2
- Document 51-9040757; Engineering Information Record: Design Requirements for Internals Handling Adapter; Revision 1

Work Orders:
- WO No. 99-3698; Polar Crane Preventative Maintenance Inspections; Revision 0
- WO No. 00-2394; NDE of reactor Vessel Internal and Lifting Facilities; Revision 0
- WO No. 200036501; Polar Crane Monthly Test; dated March 14, 2006
- WO No. 200036502; Polar Crane Monthly Test; dated April 13, 2006
- WO No. 200116889; Preventative Maintenance: EDB-SUB099-11,Containment Cranes; dated April 20, 2006
- WO No. 200116931; Preventative Maintenance: EDB-SUB062-01-001, Reactor Vessel Internal and Lifting Facilities; dated April 20, 2006
- WO No. 200143430; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143431; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143432; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143433; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143434; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143435; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143438; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200143439; Periodic Test: Procedure DB-MM-04010 - Special Lifting Devices
- WO No. 200206435; Periodic Test: Procedure DB-MM-04004 - Polar Crane Monthly Test

Drawings:
- Drawing C-150; Containment Internal Structures, Reactor Shield Wall, Sheet 1; Revision 5
- Drawing C-151; Containment Internal Structures, Reactor Shield Wall, Sheet 2; Revision 4
- Drawing C-152; Containment Internal Structures, Reactor Shield Wall (Reinf.), Sheet 3; Revision 5
- Drawing C-154; Containment Internal Structures, Reactor Shield Wall (Reinf.), Sheet 4; Revision 7
- Drawing C-175; Reactor Beam - Support Details, Sheet 1; Revision 4
- Drawing 152055; Coolant Pipe Assembly, Plan View; Revision 7
- Drawing 154613; Arrangement, Reactor Vessel Long Section; Revision 8
- Drawing 154620; Detail and Sub-Assembly, Inlet Nozzle; Revision 2

Calculations:
- Calculation 32-1151112; TED Handling Fixture; Revision 5
- Calculation C-CSS-062.01-018; Validation of Selected Parameters for Load Drop Analysis of RPV Head onto RPV; Revision 0
- Calculation C-CSS-062.01-021; Assessment of Turnbuckle 378 for RPV Head Slings; Revision 0
- Calculation C-CSS-062.01-022; Document Qualification of RPV Head Slings and Sockets; Revision 0
- Calculation C-CSS-062.01-023; RV Closure Head Fixed Lifting Pendant Analysis; Revision 0
- Calculation C-CSS-062.01-023; RV Closure Head Fixed Lifting Pendant Analysis; Revision 1
- Calculation C-CSS-062.01-023; RV Closure Head Fixed Lifting Pendant Analysis; Revision 2
- Calculation C-CSS-062.01-024; Internals Handling Adapter Analysis; Revision 0
- Calculation C-CSS-062.01-024; Internals Handling Adapter Analysis; Revision 1
- Calculation C-CSS-062.01-024; Internals Handling Adapter Analysis; Revision 2
- Calculation C-CSS-062.01-025; Reactor Vessel Head Drop Analysis; Revision 0
- Calculation C-CSS-062.01-025; Reactor Vessel Head Drop Analysis; Revision 1
- Calculation C-CSS-099.11-010; Polar Crane - RV Internals Handling Extension; Revision 0
- Commitment No. A21778; Maintenance Review of RIS 2005-25; dated December 1, 2006

Other:
- ECP 06-0128; Engineering Change Package: Reactor Vessel Head Solid Lifting Pendant;
- Revision 0
- NG-DB-00117; Administrative Procedure: Shutdown Defense In Depth Assessment; Revision 00
- NRC Letter to Toledo Edison (Log 1584), Subject: Control of Heavy Loads - NUREG-0612 Phase II (Generic Letter 80-113), Draft Technical Evaluation Report-C5506-489; dated August 22, 1984
- NRC Letter to Toledo Edison (Log 1634), Subject: Control of Heavy Loads Near Spent Fuel-Phase I (Generic Letter 80-113), Technical Evaluation Report-C5506-348; dated October 29, 1984
- PM 0830; Preventative Maintenance: Inspection of Polar Crane; Revision DB-REV-07-2722
- SDCN 30-9014944; Safety Document Change Number: Procedure 03-5016141-02; Revision 0
- SDCN 30-9015718; Safety Document Change Number: Procedure 03-5016126-003; Revision 0
- SDCN 30-9015832; Safety Document Change Number: Procedure 03-5016126-003; Revision 0
- SDCN 30-9016322; Safety Document Change Number: Procedure 03-5016141-02; Revision 1
- Toledo Edison Serial Letter 774 to NRC, Subject: Control of Heavy Loads - Phase I (Generic Letter 80-113); dated February 1, 1982

- Toledo Edison Serial Letter 952 to NRC, Subject: Control of Heavy Loads - Phase II (Generic Letter 80-113); dated June 10, 1983
- VPROC-02-0009-00; Contractor Procedure Review Form: Procedure 03-5016126-00, RV Head Removal; dated February 14, 2002
- VPROC-02-0009-01; Contractor Procedure Review Form: Procedure 30-5023155-00, SDCN for RV Head Removal; dated December 12, 2002
- VPROC-02-0009-02; Contractor Procedure Review Form: Procedure 30-5023155-00, SDCN for RV Head Removal; dated January 11, 2003
- VPROC-02-0010-00; Contractor Procedure Review Form: Procedure 03-5016141-01, RV Head Reinstallation; dated February 14, 2002
- VPROC-02-0010-01; Contractor Procedure Review Form: Procedure 03-501614101, RV Head Reinstallation - SDCN-30-5017407; dated March 25, 2002
- VPROC-02-0010-02; Contractor Procedure Review Form: Procedure 30-5017407-01, SDCN for RV Head Reinstallation; dated December 12, 2002
- VPROC-02-0010-03; Contractor Procedure Review Form: Procedure 30-5017407-02, RV Head Reinstallation; dated January 11, 2003

1R22 Surveillance Testing

Condition Reports:
- CR 07-29193; Key-Operated Hand Switch was Turned in the Wrong Direction
- CR 07-29089; NRC's Question of Exceeding EDG Fuel Filter Design Pressure Element Swap
- CR 03-01627; #2 EDG DC Fuel Oil Pump Low Discharge Pressure During Testing
- CR 02-07428; LIR: EDG Fuel Oil Filter Min/Max Level Not Adequate
- CR 03-01943; Diesel Fuel Total Particulate Level
- CR 02-06062; LIR: EDG Fuel Filter Inlet Operating Pressure Exceeds Vendor Limits for Change

Other:
- System Description for Emergency Diesel Generators and SBO Diesel Generator, Rev 5
- GMD Systems Inc. Maintenance Manual for 645E4C/F4B Turbocharged Engine, Rev 3
- Operations Tour Sheets for EDG 1, Rev 7

Procedures:
- NOBP-LP-2604-03; Maintenance Excellence Job Briefing Checklist; Rev 3
- B-ME-xxxxx xxxxxx; xxxxxxxxxxxxxxxxxx; Revision xx
- DB-MI-03011; Channel Functional Test of the Reactor Trip Breaker B, RPS Channel 1 Reactor Trip Module Logic, and ARTS Channel 1 Output Logic; Revision 16
- DB-SP-03337; Containment Spray Quarterly Pump and Valve Test; Revision 14
- Operational Schematic OS-005; Containment Spray System; Revision 11
- DB-SC-04271; Periodic Test Procedure for SBODG Monthly Test, Rev 10
- DB-MM-09343; Emergency and Station Blackout Diesel Engine 2 Year Maintenance of Lube Oil Filters, One Revolution and Other Inspections, Rev 00
- DB-OP-06316; Diesel Generator Operating Procedure, Rev 34
- DB-MM-09320; Emergency and Station Blackout Diesel Engine Maintenance, Rev 12
- DB-PF-03001; Main Steam Safety Valve Setpoint Test; Revision 03
-

Work Orders:
- WO2002234439; MI3011-01 08.000 Channel 1 Breaker B, RTM/ARTS Logic FA NORM
- WO200232981; SC4271-001 05.00 K5-03 D214 SBODG Monthly Test FA NORM

Drawings:
- Drawing Dresser CP-1006; ASME Section III Maxiflow Safety Valves; Revision dated July 30, 2003

1EP4 Emergency Action Level and Emergency Plan Changes

Other:
- Davis-Besse Nuclear Power Station Emergency Plan; Revisions 24 and 25

2PS2 Radioactive Material Processing and Transportation

Condition Reports:
- CR 07-26899; There is no defined mechanism for tracking shipper qualifications in FITS (FENOC Integrated Training System
- CR 07-26534; Radiation Protection Audit; Inspection frequency for radioactive container
- CR 07-26537; Radiation Protection Audit; Davis Besse USAR may not accurately describe current practice dealing with DAW
- CR 07-26532; Radiation Protection Audit; Radiological controls were not maintained for shipment until shipment exited the site
- CR 07-26532; Radiation Protection Audit; a Transport Index value of 0.9 was documented on the Uniform Low Level Radioactive Waste Manifest 06-2017 when correct value was 0.6

Other:
- Waste Steam Report; 10 CFR Part 61 Compliance Data Technical Basis for Davis Besse, SRST 07-04 water Processing; dated May 23, 2007
- Waste Steam Report; 10 CFR Part 61 Compliance Data Technical Basis for Davis Besse, Duratek Resin 2006; dated February 8, 2007
- Waste Steam Report; 10 CFR Part 61 Compliance Data Technical Basis for Davis Besse, Secondary Resin 0701; dated May 7, 2007
- Waste Steam Report; 10 CFR Part 61 Compliance Data Technical Basis for Davis Besse, DAW 0412; dated May 24, 2007
- Update Safety Analysis Report, Section 11; Revision 23
- GEL Laboratories LLC; Title 10 CFR 50/61 Certificate of Analysis of Davis Besse Waste Stream; dated May 14, 2007
- Davis-Besse Nuclear Power Station Process Control Program; Revision 7; Effective Date of May 9, 2005
- FENOC (First Energy Nuclear Operation Company); Self-Assessment Snap DB-SS-05-14; Improvements to Liquid Radioactive Waste Processing; dated September 21-23, 2005
- FENOC- Davis Besse 2005-2007 Waste Processor Log, Miscellaneous Shipment Log
- Uniform Low Level Radioactive Waste Manifest 07-2008; Low Specific Activity (LSA-II) shipment Resin to Studsvik Processing Facility LLC, Erwin, TN; dated June 6, 2007
- Uniform Low Level Radioactive Waste Manifest 06-3030; Surface Contaminated Object (LCO-II), AREVA equipment to Framatome, Lynchburg, VA; dated March 31, 2006
- Uniform Low Level Radioactive Waste Manifest 06-2019; Type B(U) shipment containing resin to Studsvik Processing Facility LLC, Erwin, TN; dated May 17, 2006
- Uniform Low Level Radioactive Waste Manifest 07-2015; Low Specific Activity (LSA-II) cask shipping containing HIC filters to Studsvik Processing Facility LLC, Erwin, TN; dated December 11, 2007

- Uniform Low Level Radioactive Waste Manifest 06-2012; Low Specific Activity (LSA-II) shipping containing metal; wood; plastic, rubber; dirt; insulation; asbestos and miscellaneous (DAW) to Duratek Bear Creek; Oakridge, TN; dated April 18, 2007
- Uniform Low Level Radioactive Waste Manifest 06-2012; Low Specific Activity (LSA-II) shipping containing metal; wood; plastic, rubber; dirt; insulation; asbestos and miscellaneous (DAW) to Duratek Bear Creek; Oakridge, TN; dated May 4, 2006
- System Description for Miscellaneous Liquid Radwaste System; Revision 3; dated March 14, 2005
- System Description for Boron Recovery System Clean Liquid Radwaste System; Revision 4; dated February 10, 2005
- SA 015876; QF-0406 Revision 2(ACP-117.4); Snapshot Report of Radwaste and Transportation; dated April 25, 2007
- FENOC Oversight ; Nuclear Oversight Observation Report; Radiation Protection and Radwaste Processing Program Audit; dated July 16, 2007

4OA1 Performance Indicator Verification

Other:
- Performance Indicator Data Input Sheets for MSPI Heat Removal System – Auxiliary Feedwater System; August 2006 though September 2007
- Performance Indicator Data Input Sheets for MSPI Support Cooling System – Service Water; August 2006 though September 2007
- Performance Indicator Data Input Sheets for MSPI Support Cooling System – Component Cooling Water; August 2006 though September 2007
- Performance Indicator Data Input Sheets for MSPI Residual Heat Removal System; August 2006 though September 2007

4OA2 Problem Identification and Resolution

Condition Reports:
- CR 07-25993, Inadequate SW Flow Through CCW HX#1
- CR 07-29978; Moisture In RE 4597AA Flow Meter And Sample Path

Other:
- CA-SA-07-061; Corporate Assessment of the Integrated Performance Assessment January – June 2007
- Davis-Besse Third Quarter Cognitive Trend Report for Chemistry
- Davis-Besse Third Quarter Cognitive Trend Report for Design Engineering
- Davis-Besse Third Quarter Cognitive Trend Report for Maintenance
- Davis-Besse Third Quarter Cognitive Trend Report for Operations
- Davis-Besse Third Quarter Cognitive Trend Report for Outage Management
- Davis-Besse Third Quarter Cognitive Trend Report for Plant and Reliability Engineering
- Davis-Besse Third Quarter Cognitive Trend Report for Procedure Control Unit
- Davis-Besse Third Quarter Cognitive Trend Report for Radiation Protection
- Davis-Besse Third Quarter Cognitive Trend Report for Regulatory Compliance
- Davis-Besse Third Quarter Cognitive Trend Report for Site Projects
- Davis-Besse Third Quarter Cognitive Trend Report for Technical Services Engineering
- Davis-Besse Third Quarter Cognitive Trend Report for Training Section
- Davis-Besse Third Quarter Cognitive Trend Report for Work Management
- DB-PA-07-03; Fleet Oversight Third Quarterly Performance Report
- DB-SA-07-051; Integrated Performance Assessments January to June 2007; Chemistry

- DB-SA-07-052; Integrated Performance Assessments January to June 2007; Design Engineering
- DB-SA-07-053R1; Integrated Performance Assessments January to June 2007; Emergency Response
- DB-SA-07-054; Integrated Performance Assessments January to June 2007;Maintanence
- DB-SA-07-055; Integrated Performance Assessments January to June 2007; Operations
- DB-SA-07-056; Integrated Performance Assessments January to June 2007; Outage Management
- DB-SA-07-057; Integrated Performance Assessments January to June 2007; Procedures
- DB-SA-07-058; Integrated Performance Assessments January to June 2007; Plant Engineering/Reactor Engineering
- DB-SA-07-059; Integrated Performance Assessments January to June 2007; Radiation Protection
- DB-SA-07-061R2.5; Integrated Performance Assessments January to June 2007; Regulatory Compliance and Performance Improvement
- DB-SA-07-062; Site Roll-up Integrated performance Assessment January to June 2007
- DB-SA-07-063; Integrated Performance Assessments January to June 2007; Site Projects
- DB-SA-07-064; Integrated Performance Assessments January to June 2007; Site Protection
- DB-SA-07-065R1; Integrated Performance Assessments January to June 2007; Technical Services Engineering
- DB-SA-07-066R1;Integrated Performance Assessments January to June 2007; Training DB-SA-07-067; Integrated Performance Assessments January to June 2007; Work Management

4OA3 Followup of Events and Notices of Enforcement Discretion

Condition Reports:
- CR 07-29028; Radioactive Material Found Outside the RCA
- CR07-29587; Number 1 Aux Feed Pump Governor Won't Come Off Low Speed Stop
- CR06-00313; AFW Turbine 2 Did Not Respond to Speed Control Signal From Control Room

Procedures:
- DB-OP-6261; Service Water System Operating Procedure; Revision 31
- DB-OP-02011; Heat Sink Alarm Panel 11 Annunciators; Revision 7
- NOP-OP-1004; Reactivity Management; Revision x
- DB-0180-0, Davis-Besse Design Criteria Manual, Rev 8
- NOP-CC-2004-02, Design Interface Review Checklist, Rev 7

Other:
- LER 05000346/2006-004-02, Potential Damage to Ventilation Dampers due to Design-Basis Tornado Differential Pressures Davis-Besse Nuclear Power Station, Unit No.1

4OA5 Other Activities

Condition Reports:
- CR 07-23306, "COIA-CAP-2007: CARB Chairman Delegation to Review Follow-up Action Items"
- CR 07-23297, "COIA-CAP-2007 CR 06-02588 Continued Investigation Not Returned to CARB"
- CR 07-23774, "COIA-CAP-2007: CA 07-15971-2 Closed to 500 Priority Notification that was Reject"
- CR 07-27715, "Adverse Component Health Trend – Motors – 2Q07"

- CR 07-29356, “COAI-CAP-2007: Initiating CAS to Perform Additional Cause Analysis”
- CR 07-29358, “COIA-CAP-2007: Extension Documentation for CAS Tracked by PRI 600
- Work Orders”
- CR 07-29360, “COIA-CAP-2007: Tracking Significant corrective Actions Outside the CAP Process”

LIST OF ACRONYMS USED

ANSI	American National Standards Institute
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
DAW	Dry Active Waste
DOT	Department of Transportation
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
GL	Generic Letter
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IR	Inspection Report
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OpESS	Operating Experience Smart Sample
PI	Performance Indicator
PM	Planned or Preventative Maintenance
RADWASTE	Radioactive Waste
RCA	Radiologically Controlled Area
RETS/ODCM	Radioactive Effluent Technical Specifications/Offsite Dose Calculation Manual
RPS	Radiation Protection Specialist
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SSC	Systems, Structures, and Components
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USAR	Update Safety Analysis Report
WO	Work Order