October 21, 2003

Mr. Lew W. Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC SPECIAL INSPECTION - SYSTEM HEALTH ASSURANCE FOLLOWUP - REPORT NO. 0500346/2003003(DRS)

Dear Mr. Myers:

On September 9, 2003, the NRC completed a special inspection at your Davis-Besse Nuclear Power Station. This inspection reviewed your actions to resolve Restart Checklist Item No. 5.b, associated with assuring the capability of safety significant structures, systems, and components to support safe and reliable plant operation. Specifically, this inspection focused on review of activities associated with the discovery phase of the System Health Assurance Plan (SH-DAP-5A-01) and the subsequent program for Resolution of Open Design Questions.

Since April 2002, Davis-Besse has been under the Inspection Manual Chapter 0350 process. The Davis-Besse Oversight Panel has assessed inspection findings and other performance data to determine the required level and focus of follow-up inspection activities and any other appropriate regulatory actions. Even though the Reactor Oversight Process has been suspended at Davis-Besse, it was used as guidance for conducting inspection activities and assessing findings. In addition to documenting the results of inspection activities conducted by inspectors at Davis-Besse during this time period, this inspection report, along with the results of other inspections will be used by the Davis-Besse Oversight Panel to evaluate Davis-Besse Restart Checklist Item 5.b.

No significant findings, as defined by the Reactor Oversight Program, were identified during this inspection. Based on the results of this inspection, we have concluded that the System Health Assurance Plan met its intent to review plant systems prior to restart to ensure that these systems were in a condition that would support safe and reliable plant operation and that the discovery phase of the program was conducted in a thorough and methodical manner in accordance with the procedures established for these reviews. The original stated intent of the program was to provide assurance that important plant systems were able to perform their safety functions and support plant restart and operation. In fact, what occurred was that the program identified many systems where either significant deficiencies, or a large number of deficiencies, existed such that these systems were not in a condition to support restart and operation and that corrective action was needed to restore these systems. Further, we noted that the most significant deficiencies were found in vital systems such as service water, emergency core cooling, diesel generator, and electrical distribution.

The program for Resolution of Open Design Questions, which was developed as a result of the discovery phase, had two fundamental elements. One involved determining extent of condition of the deficiencies identified during the discovery phase and the second was resolution of system deficiencies through the use of the station's established corrective action program. This inspection monitored and evaluated the extent of condition review element. We concluded that these extent of condition reviews were conducted in an appropriate manner with acceptable results. Resolution of identified deficiencies was examined by an NRC Corrective Actions Team inspection, which will be documented in another inspection report.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Christine Lipa Acting for/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

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

Enclosure: NRC Special Inspection Report
No. 50-346/03-03 (DRS)

cc w/encl: The Honorable Dennis Kucinich
G. Leidich, President - FENOC
Plant Manager
Manager - Regulatory Affairs
M. O'Reilly, FirstEnergy
Ohio State Liaison Officer
R. Owen, Ohio Department of Health
Public Utilities Commission of Ohio
President, Board of County Commissioners
Of Lucas County
Steve Arndt, President, Ottawa County Board of Commissioners
D. Lochbaum, Union Of Concerned Scientists
J. Riccio, Greenpeace
P. Gunter, Nuclear Information & Resource Service

The program for Resolution of Open Design Questions, which was developed as a result of the discovery phase, had two fundamental elements. One involved determining extent of condition of the deficiencies identified during the discovery phase and the second was resolution of system deficiencies through the use of the station's established corrective action program. This inspection monitored and evaluated the extent of condition review element. We concluded that these extent of condition reviews were conducted in an appropriate manner with acceptable results. Resolution of identified deficiencies was examined by an NRC Corrective Actions Team inspection, which will be documented in another inspection report.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Christine Lipa Acting for/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

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

Enclosure: NRC Special Inspection Report
No. 50-346/03-03 (DRS)

cc w/encl: The Honorable Dennis Kucinich
G. Leidich, President - FENOC
Plant Manager
Manager - Regulatory Affairs
M. O'Reilly, FirstEnergy
Ohio State Liaison Officer
R. Owen, Ohio Department of Health
Public Utilities Commission of Ohio
President, Board of County Commissioners
Of Lucas County
Steve Arndt, President, Ottawa County Board of Commissioners
D. Lochbaum, Union Of Concerned Scientists
J. Riccio, Greenpeace
P. Gunter, Nuclear Information & Resource Service

DOCUMENT NAME: G:DRS\ML032950012.wpd

***See Previous Concurrence**

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

OFFICE	RIII *	E	RIII *	N	RIII *		RIII	
NAME	MFarber:tr		JLara		CLipa		CLipa for JGrobe	
DATE	09/23/03		09/24/03		10/09/03		10/21/03	

OFFICIAL RECORD COPY

<u>ADAMS Distribution</u>:
AJM
DFT
SPS1
RidsNrrDipmIipb
GEG
HBC
CST1
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1
DB0350

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	0500346/2003003
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449
Dates:	January 13 through September 9, 2003
Inspectors:	M. Farber, Senior Reactor Inspector R. Daley, Reactor Inspector J. Jacobson, Senior Mechanical Engineer R. Langstaff, Senior Reactor Inspector K. Sullivan, Contractor, Brookhaven National Laboratory
Approved by:	Julio Lara, Chief Electrical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000346/2003-003(DRS); Davis-Besse Nuclear Power Station; 01/13/2003 - 09/09/2003; System Health Assurance Follow-up Inspection

The report covers a special inspection, by four regional inspectors and a contract inspector, of the Davis-Besse Nuclear Power Station System Health Assurance Building Block. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>Inspector-Identified Findings</u>

None

B. <u>Licensee-Identified Findings</u>

None

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up (93812)

.1 Background

The System Health Assurance (SHA) Plan was one of seven building blocks identified as part of the licensee's Return to Service Plan following identification of the degradation of the reactor head. The intent of the SHA plan was to review plant systems prior to restart to ensure that these systems were in a condition that would support safe and reliable plant operation. The plan consisted of three review programs: (1) an Operational Readiness Review; (2) a System Health Readiness Review (SHRR); and (3) a Latent Issues Review (LIR).

The NRC's initial inspection of the SHA plan (documented in Inspection Report 50-346/2002-013) monitored all aspects of the plan's implementation on a real-time basis, including quality assurance oversight. At the close of that inspection a limited number of SHRRs and none of the LIRs had been completed. One objective of this inspection was to review a sample of completed SHRRs and all five of the LIRs.

As part of the NRC's inspection of the SHA plan, a Safety System Design and Performance Capability Inspection of the service water, high pressure injection, and 4160 Vac electrical distribution systems was conducted. This inspection was documented in Inspection Report 50-346/2002-014. As a measure of the effectiveness of the SHRR and LIR programs, the licensee conducted independent self-assessments of the same systems. These system examinations, and the LIRs, identified numerous discrepancies involving configuration control, hardware conditions, inconsistent and potentially non-conservative assumptions in design basis and licensing basis documents, missing or unavailable calculations, and operating and test procedures which did not reflect design and licensing basis documents. These discrepancies were documented on condition reports (CR) and assessed for operability impact and significance in accordance with the station's corrective action program.

Milestone No. 19 in the Davis-Besse System Health Discovery Action Plan directed the review of SHRR and LIR reports, the identification of programmatic weaknesses or generic issues that had the potential to impact more than one system, and the performance of a collective significance review of the findings. A potential safety consequences review of approximately 600 of the more significant CRs was performed to identify trends and develop approaches to correcting the discrepancies, evaluating extent of condition, addressing the trends, and resolving design issues. As a result of these reviews a three-phase strategy was developed. This strategy was defined in a memorandum, "Resolution to System Health Assurance Plan Design Issues," and is summarized here.

- Path A - Resolution of each condition identified and determination of the extent of condition. This approach used the station's corrective action program to determine cause, extent of condition, and implement specific corrective actions for discrepancies

- Path B - Evaluation to provide additional assurance of significant safety functional capabilities. The collective review identified numerous deficiencies in the areas of calculations and testing which were intended to validate or verify the capability of safety systems to perform their functions. To more fully evaluate the capability of safety systems to perform their functions, the licensee developed the Safety Function Validation Project (SFVP), which examined calculations and testing for an additional ten safety systems.

- Path C - Resolution of Design-Related Programmatic Issues. The collective reviews identified numerous discrepancies in five design-related programmatic areas (station flooding, high energy line break, environmental qualification, seismic qualification, and Appendix R - Safe Shutdown) within each of the five LIR systems. A specific detailed examination of CRs was conducted to identify, characterize, determine extent of condition if necessary, and prescribe program or process corrective actions for problems in each of those programmatic areas.

a. Inspection Scope

NRC inspectors reviewed three SHRRs, all five LIRs, the two discovery phase collective significance review reports, the SFVP report, and the five programmatic collective significance review reports. The inspectors spent one week in detailed oversight of the SFVP, reviewing procedures, reports, and monitoring oversight panel meetings. The inspectors also reviewed related CRs, attended meetings, and interviewed members of the licensee's staff involved in system health activities.

b. Observations and Findings

b.1 Safety Function Validation Project

Introduction:

Analysis of discrepancies identified by the LIRs, the NRC Safety System Design and Performance Capability Inspection, and the three licensee self-assessments revealed that approximately eight percent of those were potentially safety-significant. Of those, design basis calculation-related discrepancies were the dominant issue. The licensee developed the SFVP to determine the extent of design basis calculation discrepancies in safety-related systems that were not subject to an LIR and to determine whether these systems could perform their accident mitigation functions. Because surveillance and other forms of system testing provide direct evidence of system functionality and discrepancies were identified in this area, testing was included in the scope of the examinations. The SFVP was expressly intended to evaluate system functionality as supported by calculation and testing. The report specifically noted that there were other aspects of system design, maintenance, and operation that could affect the ability of the systems to perform their safety functions.

C-NSA-99-16.70, "Fussell-Vesely Importance Values for Safety Related Systems," Revision 0, dated December 6, 2003, was performed to rank-order safety systems in accordance with their contribution to overall core damage frequency (CDF). This calculation determined that fifteen safety systems contributed more than 1 percent to CDF (i.e., if perfect system reliability was assumed, the CDF would decrease by more than 1 percent), and that collectively, the postulated failure of these 15 systems contributed more than 99 percent of CDF and large early release frequency. Five of these systems were examined by the LIR process; the remaining ten safety systems were examined by the SFVP. These systems are listed as follows:

Safety Features Actuation (SFAS);
Steam and Feedwater Rupture Control (SFRCS);
125/250 Volt dc Electrical Distribution (125/250 Vdc);
480 Volt ac Electrical Distribution (480 Vac);
4160 Volt ac Electrical Distribution (4160 Vac);
Main Steam (MS);
High Pressure Injection (HPI);
Steam Generators (SGs);
Emergency Core Cooling - Heating Ventilation and Air Conditioning (ECCS-HVAC); and
Decay Heat Removal/Low Pressure Injection (DHR/LPI).

The inspectors considered the justification for the scope and system selection technically acceptable.

Description:

b.1.1 Safety Function Validation Process

The SFVP was intended to validate that system safety functions would be performed by confirming that the system's design basis calculations demonstrated safety function capability, or that applicable tests were performed which demonstrated safety function capability. The emphasis of the reviews was on safety functions. For each system, the relevant sections of the Updated Safety Analysis Report (USAR), Technical Specifications, System Description, and the Design Basis Validation Project (DBVP) report were used to identify the system's safety and accident mitigation functions. The expected calculations and tests were defined based on the list of safety and accident mitigation functions and the information developed in the DBVP. The design basis calculations and testing were assessed against the requirements of the safety functions and a system-specific SFVP report was prepared for each system, documenting the results of the reviews.

b.1.2 Oversight Panel Activities

A project Oversight Panel was created to review and approve work performed by the system teams throughout the SFVP. The Oversight Panel was composed of senior engineering personnel with in-depth experience and knowledge of the design and operation of nuclear power plant systems. The responsibilities of the Oversight Panel included providing oversight and assessment of the SFVP process, ensuring the consistency and appropriate technical depth of the SFVP results, reviewing and preliminarily approving system boundaries, safety functions, and attributes, and reviewing results produced by the system teams. The Oversight Panel also provided feedback and comments to the system review teams. They also concentrated on potential safety concerns and ensured that the recommendations related to those safety concerns were adequate.

b.1.3 System-Specific Safety Function Validation Report Review

System-specific reviews for the ten systems were contained within the body of the overall report. The inspectors reviewed all ten of these reports; all ten system specific reports are listed in the Documents Reviewed section of this report's attachments.

c.1 Observations and Findings

Mechanical System Reviews:

The SFVP concluded that the MS, SG, HPI systems were capable of performing their safety functions and that the ECCS-HVAC and DHR/LPI systems could not perform all their safety functions. Three factors resulted in a conclusion that the ECCS-HVAC system was not capable of performing its safety functions. There were identified deficiencies in the determination of ECCS pump room heat loads, the use of an air temperature below the room air temperature in determining the air mass flow rate through the room coolers, and the use of a potentially non-conservative overall heat transfer coefficient for the room coolers in the most recent calculation (Calculation 12501-M-003) of peak room temperature. Prevention of in-core boron precipitation after a loss of coolant accident (LOCA) using the decay heat drop line was the only safety function that could not be proven to be satisfactorily performed by the DHR/LPI system. This was not a significant deficiency as this is the backup method of boron precipitation control.

The inspectors noted that the SFVP report concluded that the HPI system was deemed capable of performing its safety functions and questioned the licensee staff with regard to the accuracy of this conclusion. The concern with inability of the HPI pumps to pass debris from the containment sump during post-LOCA recirculation had been identified two months earlier and was under evaluation by the licensee's engineering staff while the SFVP was being conducted. Subsequent to the issuing of the SFVP report, the licensee concluded that the HPI system could not perform the function of post-LOCA recirculation and identified the need to modify or replace the existing pumps. The inspectors spoke with members of the SFVP Oversight Panel who remembered discussing this specific issue and concluding that it would be excluded from the report because the potential issue was already under evaluation through the licensee's corrective actions program, no determination had been made, and as discussed earlier, the SFVP was intended to be the extent of condition review for calculation and testing discrepancies in systems not subjected to an LIR. Considering the technical justification for the scope and system selection, that the issue had been already identified, and had been considered by the Oversight Panel, the inspectors concluded that the HPI pump condition was not indicative of a flaw or weakness in either the approach or execution of the SFVP.

Electrical System Reviews

The SFVP concluded that the SFAS could perform all required safety functions, the SFRCS could not, and the ability of the 125/250 Vdc, 480 Vac, and 4160 Vac system was indeterminate. A conservative setpoint calculation and an incorrect calibration test procedure acceptance criteria resulted in an SFRCS setpoint for differential pressure trip which might not actuate at the required analytical value. In addition, no basis could be located for the analytical value.

Three electrical systems were found to be indeterminate (that is, the scope of the SFVP reviews could not determine if the system was capable of performing all required safety functions). This was because system capability was potentially impacted by previously identified issues being addressed by the licensee. In the 125/250 Vdc system, concerns were identified with the ability of the system to maintain minimum equipment voltages and with dc bus fault protection and coordination. Concerns with the 480 Vac and 4160 Vac systems involved load capacity, voltage adequacy, fault protection, and over and undervoltage protection. The majority of those issues involved deficiencies and discrepancies in the Electrical Load Management System (ELMS) calculations. The licensee has determined that once design issues are corrected and the electrical distribution system is properly modeled, then setpoints will be calculated using the Electrical Transient Analysis Program (ETAP), a more state-of-the-art program that is being successfully used by other utilities.

From this inspection, it was evident that the design basis health of the Davis-Besse electrical distribution systems is dependent upon the successful completion of the loading and running of the ETAP software. There are multiple issues and CRs that are open pertaining to the old electrical distribution software - ELMS. The licensee has expressed their intent to make all necessary changes and incorporate corrective actions into the new software - ETAP. Once the ETAP software is fully loaded and successfully run, the system health of the Davis-Besse electrical distribution system can be determined.

General Conclusions:

The inspectors concluded that the individual system examinations were methodical, rigorous, and thorough. The process was conservative and well-defined, the SFVP teams conducted their examinations carefully, and the oversight panel provided guidance, intense review, and good technical feedback. System boundaries and safety functions were conservatively established and defined to include, rather than exclude, components. Issues and concerns were captured in the licensee's corrective action program by promptly writing CRs.

b.2 System Health Readiness Reviews

The purpose of the SHRR was to provide reasonable assurance the system under review was capable of performing its risk-significant functions as specified by the station's Maintenance Rule program. SHRRs were conducted in accordance with station procedure EN-DP-01504, "System Health Readiness Review," up through Revision 04. These reviews were a "vertical slice" evaluation with a focus on key components. Key components were selected based on risk significance, maintenance history, and industry experience. The reviews examined testing, maintenance, modifications, corrective actions, and commitments related to the systems; design basis was not evaluated by these reviews. The reports detailed the system boundaries, the bases for those boundaries, and the related system interfaces. All of the reports contained detailed lists of all documents reviewed, walkdown results, and CRs initiated by the review teams.

b.2.1 Decay Heat Removal/Low Pressure Injection

The decay heat removal/low pressure injection (DH/LPI) SHRR evaluated both risk-significant and non-risk significant maintenance rule functions. Focus components included a DH pump, the borated water storage tank, a flow nozzle, a DH cooler, and eight critical valves.

The SHRR concluded that there was not reasonable assurance that the system could perform its risk-significant functions and was thus not ready to support restart. The review identified several weaknesses in testing, the need to resolve issues from industry experience, modifications, and commitments, and the need to correct a large number of deficiencies in materiel condition. All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

b.2.2 Station Blackout Diesel Generator (SBODG) and SBODG Heating, Ventilation, and Air Conditioning (HVAC) System

The SBODG and SBODG-HVAC SHRR evaluated the risk-significant maintenance rule function of providing on-site standby power to both safety and nonsafety-related equipment in the event of a station blackout. Focus components included a pressure regulating valve, starting air compressor, soakback pump, engine governor, and the air start compressor supply breaker.

The SHRR concluded that the SBODG was not ready to support plant restart. The most significant issue was that the re-crank circuit for the air start motors would not protect against pinion gear abutment against the starter gear on the engine flywheel. Resolution of this deficiency involves relocating the pressure tap for the re-crank pressure switches and changing the pressure switch setpoints. Other deficiencies included weaknesses in testing, need to upgrade the starting air system, and replacement of the governor. All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

b.2.3 Containment System

The containment system SHRR evaluated the risk-significant maintenance rule function of containment isolation to ensure that dose was maintained below 10 CFR Part 100 limits. To ensure that the review adequately addressed the function, containment isolation components from systems not undergoing a latent issues review were incorporated into the containment system. The list of systems included, but was not limited to, core flood system, steam generators, refueling canal and transfer tube, nitrogen purge and blanket system, reactor sample system, and containment vessel gas monitoring, purge, hydrogen dilution, and spray systems. The containment SHRR specifically excluded the opening cut in the containment to replace the reactor head. Restoration and validation of containment integrity associated with this effort were addressed by the Integrated Leak Rate Test which was discussed in inspection report 50-346/03-05(DRS). The actual structure of containment was addressed by the Containment Health Assurance Plan. Focus components included containment purge isolation valves, two electrical penetrations, the personnel airlock, containment spray isolation valves, and the containment vacuum breaker isolation valves.

The SHRR concluded that the containment system was not ready to support plant restart. The primary restraint was the need to address material condition deficiencies, the most significant of which were boric acid residue, corrosion on and in containment penetrations, and peeling coatings on the upper containment vessel (dome) and containment spray supports. All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

c.2 Observations and Findings

After reviewing the three SHRRs discussed above and the two discussed in inspection report 50-346/02-13(DRS), the inspectors concluded that the SHRRs were conducted in accordance with the station's procedure, that the reviews were detailed and intrusive, and that the conclusions were conservative and well-supported.

The inspectors questioned whether the re-crank circuit deficiency with the SBODGs had been identified and was being addressed for the station's emergency diesel generators (EDG) since they were essentially the same engine. The licensee stated that this was the case. The inspectors verified this during the review of the EDG Latent Issues Review report which showed that Engineering Work Request 01-0402-00 was the corrective action.

b.3 Latent Issues Reviews

NRC reviews of the purpose, process, and procedure for Latent Issues Reviews (LIRs) were discussed in inspection report 50-346/02-13(DRS). The LIRs were a concentrated effort that evaluated the systems' capability to perform safety and accident mitigation functions and to support continued safe and reliable operation. The process used to perform the LIR was a "vertical-slice" approach, similar to the NRC procedure used to perform the historic Safety System Functional Inspections and the current Safety System Design and Performance Capability Inspections. The LIRs consisted of extensive multi-disciplined documentation reviews, system walkdowns, and plant personnel interviews. During this inspection, the completed LIR reports for the five selected systems were examined to determine whether the reviews had been conducted in accordance with the LIR procedure, and to evaluate the thoroughness of the review and the reports' conclusions on the functional capabilities of the five systems. The LIR collective significance review report was examined to evaluate the report's conclusions on potential generic issues.

b.3.1 <u>Service Water</u>

The service water LIR evaluated the following service water system and ultimate heat sink safety and accident mitigation functions:

- The safety function of the ultimate heat sink is to provide sufficient cooling water for the service water system for at least 30 days. This supply of water is provided by Lake Erie and maintained in and adjacent to the intake structure forebay.
- Provide a safety-related cooling water supply to the component cooling water heat exchangers during normal and emergency plant operation.
- Provide a safety-related cooling water supply to the containment air coolers during normal and emergency plant operation.
- Provide a safety-related cooling water supply to the emergency core cooling system room cooling coils during normal and emergency plant operation.
- Provide a safety-related cooling water supply to the control room emergency ventilation system and hydrogen dilution blowers during emergency plant operation.
- Provide a backup safety-related cooling water supply to the auxiliary feedwater system, component cooling water system, and motor-driven feedwater pump during emergency plant operation.

The LIR concluded that the system was not ready to support restart as a result of a number of significant discrepancies which seriously impacted the capability to perform its safety and accident mitigation functions. Some of the more significant discrepancies included:

- Lack of setpoint calculations for instrumentation setpoints;
- Flawed and non-conservative analyses for the ultimate heat sink, heat transfer analyses for the ECCS room coolers, and acceptance criteria for the flow balance test;
- Failure of the non-seismic cooling tower makeup pumps or piping located in the service water pump room would result in flooding of the room within minutes; and
- Deficiencies in analyses for the electrical supply to the service water pump motors.

All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

b.3.2 Emergency Diesel Generator

This LIR evaluated whether the emergency diesel generator (EDG) system was capable of meeting its safety and accident mitigation function of providing standby power for safety-related loads required to mitigate the effects of a design basis accident combined with a loss of offsite power and to safely shut down the plant and maintain the plant in a safe shutdown condition. The scope of the EDG LIR included the diesel engine, generator, and related support systems.

The LIR concluded that the EDGs were not ready to support restart as a result of a number of significant discrepancies which seriously impacted the capability to perform its safety and accident mitigation functions. Some of the more significant discrepancies included:

- Calculations associated with support of operation with ambient air temperature above 95 degrees Fahrenheit or with diesel room temperature above 120 degrees Fahrenheit, incorporation of new loads and verification of voltage and frequency during the loading sequence, and support of seismic requirements for the underground diesel fuel oil piping were either flawed or missing;

- Inadequate missile protection for the diesel exhaust stack, corrosion and minimum wall thickness concerns for EDG 1-2 exhaust piping, automatic voltage and frequency control upon receipt of a diesel start signal, and noncompliance with a General Electric Service Information Letter on ensuring the seismic capability of GE HFA Relays;

- Material condition issues with flex hoses, fuse oxidation, fuel oil leaks, and bolting; and

- Procedure inadequacies in both operations and surveillance test procedures.

All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

b.3.3 Component Cooling Water

The component cooling water (CCW) LIR evaluated the following safety and accident mitigation functions:

- Removing the maximum applied heat loads from safety-related components during design basis events with service water at the ultimate heat sink temperature;

- Under post-LOCA conditions, where the backup boron precipitation control method is required, procedurally controlled actions are credited with maintaining the CCW inlet temperature to the decay heat removal cooler at a minimum of 95 degrees Fahrenheit, prior to commencing this operation;

- Section 6.2.4.2 of the USAR requires that piping penetrations for lines that do not normally connect directly to the containment atmosphere, but may fail following a seismic event, be classified Type II. This consideration applies to the CCW penetrations (i.e., Nos. P3, P4 and P12). The Type II isolation scheme for the CCW lines requires one automatic isolation valve inside and one automatic isolation valve outside containment. Check valves may not be used outside containment as isolation valves; and

- Supporting makeup pump operation during emergency feed-and-bleed operations following a total loss of feedwater (non-design basis accident conditions). Crossties to the essential CCW header are required to permit isolation of the nonessential headers, without loss of cooling to the makeup pump gear and lube oil coolers.

The LIR concluded that there was insufficient assurance that the system was capable of satisfying the safety and accident mitigation functions during operational modes one through four. Some of the more significant discrepancies included:

- Removing the design heat load from the DHR coolers, LPI and HPI pump coolers and containment gas analyzers at the maximum postulated service water interface temperature;

- Ensuring the functional capability of the CCW surge tank;

- Ensuring the standby pump is capable of supporting the operation of the associated emergency diesel generator during a loss of offsite power without a loss of coolant accident;

- Maintaining the integrity of the primary containment following design basis accidents;

- High energy line break concerns related to turbine building line breaks that could affect the CCW pump room barriers and the operating environment;

- Lack of documentation to verify the basis for the current field settings for the CCW System instruments and no setpoint/uncertainty calculations for the safety-related non-Technical Specification instrumentation in the system;

- Lack of documentation (motor curve, etc.) to substantiate that the CCW pump motors were capable of starting at 70 percent of the nominal voltage rating;

- Inability to validate the maximum accident heat removal rate for the DHR heat exchanger of 1.10E8 BTU/hr in USAR Table 9.2-7;

- Lack of functional testing of interlocks installed for the letdown coolers and reactor coolant pump seal coolers to prevent overpressurization of the CCW piping in the event of a tube rupture;

- Anomalies in evaluations of piping, valves and pressure retaining components within the safety-related boundary of the system; and

- Lack of documentation to verify the adequacy of the steel grating being credited with protecting the CCW pump room HVAC louvers and exposed equipment from external missiles.

All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

b.3.4 Auxiliary Feedwater

The auxiliary feedwater (AFW) LIR evaluated the following safety and accident mitigation functions:

- Provide an emergency source of safety-related feedwater to secondary side of the steam generators;

- Remove decay heat from the reactor coolant system (RCS) to prevent over pressurization of that system on design basis accidents;

- Promote driving head for natural circulation responding to a steam feed rupture control system signal, in cases where all reactor coolant pumps are unavailable including a loss of offsite power; and

- Limit cooldown of RCS to maintain shutdown margin, RCS integrity, and containment integrity

The LIR also evaluated a number of other functions which were not considered safety and accident mitigation. The inspectors reviewed these functions and concurred with this determination.

The LIR determined that there was reasonable assurance that the AFW system was able to perform its safety and accident mitigating functions and provide reliable future operation, but that significant issues identified in the report needed to be resolved prior to plant restart. These issues included:

- environmental qualification program deficiencies;

- high energy line break program deficiencies;

- seismic calculation deficiencies;

- potential common-mode failure due to a blockage of service water flow to the turbine bearing coolers; and

- ambiguity in the licensing basis regarding the required suction source for AFW during design basis accidents

All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed. The inspectors noted that the programmatic deficiencies were common to other systems and that collective significance reviews (discussed in section b.4 below) were conducted for these programs.

b.3.5 Reactor Coolant

The RCS LIR evaluated the following safety and accident mitigation functions:

- Transfer heat from the core to the steam generator during steady state operation and for design transients without exceeding core thermal limits;
- Remove decay heat from the core via redundant components and features using controls from inside or outside the control room;
- Provide for natural circulation cooldown from normal operating temperature and pressure to conditions that permit operation of the decay heat removal system;
- Form a barrier against the release of reactor coolant and radioactive material to the environment; and
- Transfer heat from the reactor core to containment during a loss of steam generator cooling with high RCS pressure using make-up/high pressure injection core cooling.

The RCS LIR concluded that the system could not support its safety and accident mitigation functions because of the following issues:

- Potential challenges to RCS safety functions caused by not applying instrument uncertainty to critical non-safety limit parameters such as pressurizer level;
- Error limits for RCS flow appear to exceed the calculated instrument uncertainty;
- The tolerances applied to acceptance criteria in the monthly channel checks for post accident monitoring and auxiliary shutdown panel did not have a documented basis and did not appear to be consistently conservative;
- The reactor pressure vessel closure head required replacement due to the degradation caused by the boric acid corrosion;
- Successful resolution of the Framatome preliminary safety concern related to potential RCS leakage from the incore instrument nozzles;
- The reactor coolant pumps had a history of casing-to-cover joint leakage; continued operation with degraded gaskets could lead to reactor coolant pressure boundary structural damage due to boric acid wastage of carbon steel components; and

- Replacement of pressurizer spray valve, RC2, which had a history of packing leakage that resulted in a licensee event report.

All of these issues were carefully documented in the report; CRs were issued to ensure that resolution was completed.

b.3.6 Discovery Phase Collective Significance-Latent Issues Reviews and Associated Inspections

Corrective Action 2 for CR 02-09224, implemented the completion of the collective significance review and the preparation of collective significance review reports to evaluate the findings from the LIR reports as directed by Milestone No. 19 in the System Health Assurance Discovery Action Plan. The Discovery Phase Collective Significance Latent Issues Reviews report was approved on December 10, 2002.

The inspectors examined this collective significance review of the discrepancies identified during LIRs and other associated inspections (NRC Safety System Design and Performance Capability Inspection report 50-346/02-14 and self-assessments performed by the licensee on the HPI, service water, and 4160 Vac electrical distribution systems). The purpose of the review was to identify programmatic weaknesses or generic issues that had the potential to impact more than one system. Information used in this review included discrepancies identified in the five LIRs and the associated inspections. Discrepancies were sorted by topical area and common attributes. The discrepancies assigned to each topic or common attribute were examined collectively to identify recurring issues, issues which could impact other systems, or which could signal a programmatic weakness. Objective criteria: (1) total number of discrepancies greater than 25; (2) greater than 5 percent of total number of checks in the given topical area resulted in a deficiency, were established for determining which of the 31 LIR topics and 13 common attributes should be considered potential generic issues. A total of 20 potential generic issues was identified. The inspectors focused their examination on those topics and common attributes that were determined not to be potential generic issues.

c.3 Observations and Findings

No findings were identified. The inspectors determined that the five LIR reviews were conducted in strict compliance with the procedure, that the LIRs were thorough and intrusive, deficiencies were accurately characterized, conclusions on the functional capability were correct and well-supported.

With regard to the discovery phase collective significance review for LIR and associated inspections, the inspectors raised the following concerns:

- Topic No. 28, Corrective Actions, was determined not to be a potential generic issue. This was an erroneous conclusion because previous NRC inspections and the licensee's management and human performance root cause had concluded that corrective action failures played a significant role in the degradation of the reactor head. The inspectors questioned how valid conclusions of “no potential generic impact,” for other topics could be if application of the criteria resulted in an erroneous conclusion of that nature.

The licensee determined that the reason was a combination of the statistical methodology used to evaluate the deficiencies, that the population of corrective actions examined by the latent issues teams were related only to equipment problems with the specific system under examination, that the teams were directed not to raise additional examples of already identified generic problem areas as there was little to be gained from that unless the specific issue was safety-significant. Consequently, the reported number of corrective action deficiencies was small (11) and when compared to the total number of checks (649), this resulted in a small percentage (1.7 percent). Since neither number exceeded the objective criteria of 25 deficiencies nor 5 percent, the topical area was not flagged as a potential generic area. The primary contributor to this was the selection criteria for the corrective action deficiencies. The inspectors considered this reasonable.

- Six of the 13 common attributes evaluated were determined to have potential generic impact: equipment qualification; seismic qualification; Appendix R; high energy line break; temperature effects on operability; and external hazards. In general, these corresponded to the five topical issues selected for Path C. The inspectors questioned why there were no collective significance reviews prescribed for external hazards and temperature effects on operability. The inspectors also questioned why station flooding was selected for a collective significance review since there was no explicit mention of flooding in the report.

  The licensee responded that it had not been intended that the common attributes from the LIR CSR correlate precisely on a one-to-one basis with the Path C topical areas. The Path C topical areas were based on potential programmatic issues from the LIR program that were not in the scope of the Design Basis Validation Project as was the case with HELB, EQ, seismic, Appendix R - safe shutdown, and flooding. The inspectors considered this acceptable.

- The inspectors noted that the report listed five problem areas in the seismic qualification common attribute, yet none of these were addressed in the Path C seismic CSR. The licensee responded that the two reports had a different focus and that in the time that elapsed between the two reports, the issues had been further evaluated and in some cases resolved. The LIR CSR was a preliminary screening while the Path C CSR was a more in-depth evaluation by seismic experts, therefore it was realistic for the two reports to have some differing results. The inspectors considered this acceptable.

b.4 <u>Design-Related Programmatic Issues</u>

The Discovery Phase Collective Significance Report identified five design-related programmatic areas that had the potential to impact systems beyond those examined in the LIR program. Plans were then developed to determine and address the extent of condition of the following design-related areas:

- High Energy Line Break (HELB);
- Station Flooding;
- Seismic Qualification;
- Environmental Qualification (EQ); and
- Appendix R - Safe Shutdown.

Appendix A in the memorandum, “Resolution of System Health Assurance Plan Design Issues,” detailed the plans for resolving each of these areas. Using the corporate guideline, collective significance reviews (CSR) were performed for each of these areas under an implementing CR and a report, which documented the findings and conclusions, was prepared for each topical area.

The inspectors evaluated the scope, depth, and thoroughness of the five CSR reports, reviewed the implementing CRs, sampled CRs associated with the topical area, and evaluated the findings, conclusions, and corrective actions specified by the licensee.

b.4.1 <u>High Energy Line Break CR 02-05526</u>

The HELB CSR examined 83 CRs from 2001 through January 2003. Each CR was evaluated and assigned to one of three categories:

- Evaluations associated with Turbine Building HELBs;
- HELB Calculation issues; and
- Non-program issues.

The non-program issues were further categorized:

- Door maintenance or labeling concerns;
- Environmental qualification concerns;
- Miscellaneous non-program concerns;
- Barrier/seal maintenance or documentation issues; and
- Flooding concerns (note: this CR was included in the flooding CSR).

The CRs were tabulated by category and then by sequence number with a short description of the associated issue. The inspectors reviewed all the issues and sampled the 83 CRs, focusing on those determined to be non-program issues. The intent of this review was to determine whether any CRs had been mis-characterized or whether additional potential categories existed. The inspectors also examined CRs to ensure that corrective actions were appropriately specified where the CSR identified an issue and recommended corrective actions. The criteria for evaluating appropriateness of corrective actions included extent of condition, accuracy in addressing the issue, and timeliness.

The HELB CSR identified four areas in need of corrective action or enhancement:

- Plant changes associated with the re-evaluation of Turbine Building HELBs;
- Correction of HELB calculations where inappropriate stress threshold values were used;
- Development of a new Auxiliary Building HELB model;
- Program enhancements to the Auxiliary Building HELB analyses; and
- Evaluation of time-critical operator actions and their bases.

c.4.1 Observations and Findings

No findings were identified. A search of the database did not reveal any CRs missed by the licensee in conducting this review. The inspectors determined that the categories were appropriate, CRs were correctly assigned to categories, issues were thoroughly evaluated, and that appropriate corrective actions or enhancements were specified.

b.4.2 Station Flooding CR 03-01517

The Station Flooding CSR examined 148 CRs from 2001 through January 2003. Each CR was evaluated and assigned to one of six categories:

- Invalid concerns;
- Administrative issues;
- Barrier issues;
- Flood source evaluation issues;
- Procedure issues; and
- Non-program issues

The station flooding CSR identified issues related to sections of seismic class II fluid systems that were upgraded to seismic class I, concerns that removal of pumps in the Service Water pump room may raise vulnerability to flooding, watertightness of rooms 51 and 52, functionality of floor drains, and need for a formal program for barriers having a function other than fire protection. The following corrective actions were specified to address these issues:

- Review specific descriptions of upgraded sections of fluid systems for accuracy;

- An evaluation was performed to determine why the cooling tower makeup pumps were removed without consideration of flooding and several corrective actions were specified;

- Conduits penetrating the external walls of rooms 51, 52, and 53 were to have flood seals installed;

- Verification of floor drain functionality was performed; and

- An evaluation of the need for a formal program for barriers having a function other than fire protection was specified.

The CRs were tabulated and the inspectors reviewed the CSR report and the associated CRs in the same manner as described in the previous CSR discussions.

c.4.2 Observations and Findings

No findings were identified. A search of the database did not reveal any CRs missed by the licensee in conducting this review.

The inspectors' review of the issues and limited sampling of CRs revealed a number of events where maintenance activities either caused some degree of flooding or had the potential to result in flooding should a barrier be inadvertently or improperly breached. Discussions with the licensee revealed that the station tagout procedure and the Shutdown Safety Procedure, which had been recently upgraded, provided substantive flood control measures. However, the aspect of on-line flood control measures had not been similarly addressed, especially for high risk activities such as water box cleaning. Subsequently, the licensee issued CR 03-04892 and CA#1 to incorporate the potential for flooding due to maintenance into NG-DB-0800, Production/Generator Risk Determination.

With the exception of the need to consider risk of flooding due to on-line maintenance, the inspectors concluded that the categories were appropriate, CRs were correctly assigned to categories, issues were thoroughly evaluated, and that appropriate corrective actions or enhancements were specified.

b.4.3 Seismic Qualification CR 03-00231

The seismic qualification CSR examined 184 CRs from 2001 through January 2003. The method used in this CSR for categorizing the CRs differed from that used in the other four. In the other reviews, the total population of CRs associated with the topical issue were assembled and then the CRs were evaluated to identify the categories. In this review, an initial population of 17 CRs was evaluated for common attributes and four categories were specified:

- Potential seismic II/I concerns;

- Apparent conflicts between seismic and quality classifications of equipment;
- Concerns with the qualification of equipment due to documentation issues or the adequacy of preventive maintenance monitoring of installed relays; and
- Non-program issues.

A subsequent search of the CR database produced an additional 167 related CRs. These additional CRs were evaluated against the previously established categories; if they did not fit one of the first three, they were considered non-program. The CSR contained a statement that no additional categories were identified. However, it was not clear on what this statement was based. The CRs were tabulated and the inspectors reviewed the CSR report and the associated CRs in the same manner as described in the previous CSR discussions.

The seismic qualification CSR identified issues related to sections of seismic class II fluid systems that were upgraded to seismic class I (same issue as identified in the flooding CSR), HFA relay chatter, pressure retaining qualifications of instruments, improper qualification of instruments, and seismic control of temporary and portable equipment. The following corrective actions were taken to address these issues or enhance the seismic qualification program:

- Review specific descriptions of upgraded sections of fluid systems for accuracy;
- Test and adust, as needed, HFA relays;
- Replace pressure instruments with improper pressure retaining qualifications and evaluate additional instruments for pressure retaining capability during normal operation and a seismic event;
- Establish clear conventions for documenting the quality or seismic classifications of equipment;
- Revise procedures for the seismic control of portable or temporary equipment;
- Develop a Seismic Program Summary to reference seismic program design documents and outline the approach to seismic qualification of station components; and
- Implement the Seismic Qualification User’s Group program for qualification of new and replacement equipment.

c.4.3 Observations and Findings

No findings were identified. A search of the database did not reveal any CRs missed by the licensee in conducting this review.

The inspectors considered the method used for identifying common attributes and determining categories less than optimum. The CSR contained nothing that showed that all 184 CRs were evaluated for common attributes; common attributes were only derived from the initial 17 CRs. The inspectors also noted that this method was inconsistent with that used in the other four CSRs. The inspectors' review of the entire CR population revealed four potential new categories. A CR (03-04924) was written to evaluate this condition and the four potential new categories. The inspectors reviewed the CR investigation and identified discrepancies in the information provided. This was discussed with licensee management, who concluded that it was necessary to have the 184 CRs reanalyzed. Correction action #4 for CR 03-00231 was issued to conduct this reanalysis, which was completed on August 27, 2003. The analysis was conducted in accordance with the licensee's corporate guidance for collective significance reviews, NOBP-LP-2006, Collective Significance Review, and identified ten categories which warranted evaluation to determine whether or not they were characteristic of programmatic problems with the Seismic Program. The review concluded that the additional categories either did not reflect programmatic problem areas or were appropriately addressed by the existing corrective actions from the original seismic CSR. After carefully reviewing the reanalysis, the inspectors concluded that it was acceptable. The need to perform a reanalysis was caused by weakness in engineering which failed to recognize the narrowness of the original method and that it was inconsistent with the other four CSRs.

b.4.4 <u>Environmental Qualification CR 03-00656</u>

The EQ CSR examined 475 CRs from 2002 through January 2003. Each CR was evaluated and assigned to one of six categories:

- Invalid Concerns - CRs where the assessment and cause analysis determined that the issue presented was not a deficiency;
- Administrative issues - CRs procedure or documentation improvements;
- EQ Programmatic Deficiencies - CRs identifying issues which could affect multiple aspects of the EQ Program;
- New or Changed Analyses Affecting Environmental Parameters - CRs addressing reanalysis of accidents;
- Qualification of Equipment Affected - CRs which addressed errors that potentially impacted qualification of components; and
- Not EQ Program Related - CRs addressing issues which when examined in depth were unrelated to the EQ program

The EQ CSR identified issues related to reanalysis of containment loss of coolant accidents and turbine and auxiliary building HELB, deficiencies with Raychem splices, drainage of EQ components, lack of understanding of EQ requirements. The following actions were specified to address these issues:

- Reevaluate EQ status for equipment that the containment loss of coolant accidents and turbine and auxiliary building HELB analyses reveal are in more severe environments than previously assumed;

- Correct Raychem splice deficiencies on level transmitters;

- Replace unqualified splices on Limitorque motor actuators;

- Corrective equipment drainage deficiencies;

- Revise the EQ program procedure, adding guidance to ensure recognition of EQ deficiencies; and

- Develop and conduct periodic EQ refresher training.

The CRs were tabulated and the inspectors reviewed the CSR report and the associated CRs in the same manner as described in the previous CSR discussions.

c.4.4 Observations and Findings

No findings were identified. A search of the database did not reveal any CRs missed by the licensee in conducting this review.

As a result of questions regarding the watertightness of rooms 51 and 52, CR 02-07782 was written to question the effectiveness of provisions in Core Drill/Cut Out and Barrier Penetration Procedure, EN-DP-01142, for watertightness of conduit fittings, lateral bends, and junction boxes. Examples where these provisions could be compromised were provided. The CR further noted that the implications of these questions extended beyond conditions in rooms 51 and 52. The inspectors reviewed the cause analysis and the specified corrective actions and noted that, in spite of the extent of condition concern raised in the CR, the apparent cause and the associated corrective actions were focused exclusively on rooms 51 and 52. The inspectors discussed this with the licensee and CA-4 to CR 02-07782, requiring an extent of condition evaluation of conduit fittings, lateral bends, and junction boxes in other rooms subject to flooding, was issued.

Based on review of the issues as described in the table and limited sampling of EQ-related CRs, the inspectors selected 23 CRs and questioned whether two additional categories: EQ administrative issues and EQ calculation issues. The licensee reevaluated these CRs and determined that additional corrective actions or enhancements were not necessary. After review of this reevaluation, the inspectors had no further concerns. The inspectors concluded that the categories were appropriate, CRs were correctly assigned to categories, issues were thoroughly evaluated, and that appropriate corrective actions or enhancements were specified. The failure to specify a corrective action to inspect for watertightness of rooms other than 51 and 52 was considered a minor violation of 10 CFR Part 50, Appendix B, Criterion XVI, that is not subject to enforcement action in accordance with Section IV of the USNRC Enforcement Policy.

b.4.5 Appendix R: Safe Shutdown CR 03-00179

The inspectors performed an independent review of safe shutdown capability to assess the quality of the Appendix R - Safe Shutdown Analysis, Collective Significance Review.

The Appendix R - Safe Shutdown CSR examined 281 CRs from 2002 through January 2003. Each CR was evaluated and assigned to one of six categories:

- Calculation/analysis;
- Documentation and procedures;
- Emergency lighting;
- Barrier and door;
- Administrative and other miscellaneous; and
- Fire protection features.

The Appendix R - Safe Shutdown CSR identified issues in all six areas, the most significant being related to lack of documented basis to support evaluations in the Fire Hazards Analysis Report, weaknesses in transient analyses, hydraulic analyses, and appendix R loading of the diesel generators, and safe shutdown procedure deficiencies. The following actions were specified to address these issues:

- The licensee’s plant vendor, Framatome, was commissioned to perform a rebaselining of the Appendix R transient analyses;
- Flow models for the service water and component cooling water system, including operation during Appendix R scenarios, were to be completed prior to entry into Mode 4;
- A bounding calculation of diesel loading during Appendix R scenarios was to be completed prior to entry into Mode 4;
- The Fire Hazards Analysis Report will be revised to provide a documented basis for the evaluations; and
- Safe shutdown procedures will be revised as necessary prior to entry into Mode 2, to incorporate information from the various analyses; operators will be trained on these revisions.

c.4.5 Observations and Findings

The inspectors concluded that the categories were appropriate, CRs were correctly assigned to categories, issues were thoroughly evaluated, and that appropriate corrective actions or enhancements were specified.

b.5 Fire Protection Features and Safe Shutdown Capability:

The team reviewed safe shutdown capability and a sample of fire protection features.

b.5.1 Transient Analyses Used to Demonstrate Safe Shutdown Capability

The team had planned on assessing the licensee's 10 CFR Part 50, Appendix R, transient analyses used to demonstrate safe shutdown capability.

c.5.1 Observations and Findings

The team was initially unable to perform a review of this area. At the time of the on-site portion of this part of the inspection (April 21, 2003 through April 25, 2003), the licensee had not yet completed updating the transient analyses used to support safe shutdown. The team noted that although the licensee did have analyses to support safe shutdown, a number of the issues had been identified with respect to the existing analyses. The licensee effort to update the analyses was expected to resolve identified issues associated with the existing analyses. During an electrical issues Inspection conducted July 7, 2003 through July 18, 2003, the new thermal-hydraulic calculations, as well as the most bounding Appendix R operations procedure, "Serious Control Room Fire," were examined. Based upon this inspection, the thermal-hydraulic calculations appeared to be comprehensive and adequate for their application at Davis-Besse. Based upon reviewing a sample of assumptions and conclusions from the thermal-hydraulic calculations, the inspection team determined that the "Serious Control Room Fire" procedure adequately incorporated the bounding conditions of the thermal hydraulic calculations.

b.5.2 Operator Actions to Accomplish Safe Shutdown

The team reviewed operator actions to accomplish safe shutdown in the event of fire and associated procedural guidance contained in procedure DB-OP-02519, Serious Control Room Fire. The team observed a simulator exercise to evaluate operator control room actions and decision making ability associated with a fire scenario requiring shutdown from outside the control room. The team also observed simulated operator actions outside the control room for required safe shutdown actions necessary to put the plant in a stable condition in the event of fire requiring safe shutdown from outside the control room.

c.5.2 Findings and Observations

As part of a simulator exercise, the team observed an operations crew exercise decision making for when to enter alternative shutdown procedures (i.e., perform a shutdown from outside the control room) in the event of a cable spreading room fire. The crew conservatively chose to enter the alternative shutdown procedure when multiple simulated indications of a cable spreading room fire and instrumentation failures existed. The team considered the crew's decision appropriate given the simulated conditions.

The team also observed the crew's in-plant simulated actions for the initial alternative shutdown actions. The team determined that the actions specified by the alternative shutdown procedure could be accomplished in a timely manner to support safe shutdown.

b.5.3 Fire Protection Features

The team reviewed fire protection features associated with one fire area, Fire Area Y, which included a high-voltage switchgear room, an electrical isolation room, and a battery room for one division.

c.5.3 Observations and Findings

The team determined that the fire protection features for Fire Area Y met 10 CFR Part 50, Appendix R, Section III.G.2 requirements and license basis requirements. Although placement of smoke detectors was approved in an NRC Safety Evaluation Report (SER), the inspectors identified that room 429A, an electrical isolation room, did not have a smoke detector. The team noted that the ventilation flow path for the room was such that air was supplied through an open door way and exhausted through an exhaust ventilation duct. As such, a fire in room 429A would likely not be readily detected because any smoke generated would tend to be exhausted through the ventilation system. Although the ventilation system did contain a smoke detector in the exhaust ducting, air sensed by the smoke detector installed in the exhaust ducting came from multiple locations and would be diluted. As such, operations personnel would have a delayed indication of a fire and would not be able to readily ascertain the location of the fire. Although they were in compliance with the SER, the licensee documented this issue on CR 03-03152 and indicated that they planned to install a smoke detector in the room as a future enhancement.

4OA6 Management Meetings

.1 Exit Meeting Summary

- The inspectors presented the inspection results to Mr. L. Myers and other members of licensee management and staff at the conclusion of the inspection on September 9, 2003. The licensee acknowledged the information presented.

KEY POINTS OF CONTACT

Licensee

K. Baker, MPR Associates
R. Coward, MPR Associates
J. Grabnar, Design Basis Engineering Manager
E. Grindahl, Nuclear Oversight
D. Gudger, Regulatory Affairs Supervisor
C. Hawley, Project Management
R. Hovland, Plant Engineering
S. Loehlein, Nuclear Quality Assurance Manager
W. Marini, Regulatory Affairs
E. Matranga, Plant Engineering
L. Myers, Chief Operating Officer
K. Ostransky, Regulatory Affairs Manager
J. Powers, Engineering Director
C. Price, Business Manager
M. Roder, Operations Manager
R. Schrauder, Support Services Director
G. Skillman, Consultant
L. Strauss, Regulatory Affairs
D. Strawson, MPR Associates
D. Studley, Engineering Assurance Board
R. Trench, MPR Associates
J. Waal, Regulatory Affairs
S. Wise, Operations Superintendent
G. Wolf, Regulatory Affairs
A. Zarechnak, MPR Associates
K. Zellers, Engineering

Nuclear Regulatory Commission

C. Lipa, Chief, Reactor Projects Branch 4
S. Thomas, Senior Resident Inspector

LIST OF DOCUMENTS REVIEWED

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

Procedures

EN-DP-01504	System Health Readiness Review; Revision 04; dated December 6, 2002
EN-DP-01505	Latent Issues Reviews; Revision 06; dated December 20, 2002
--	Davis-Besse System Health Assurance Plan; Revision 5; dated December 6, 2002
--	System Health Assurance Discovery Action Plan; Revision 2; dated October 15, 2002
--	Desktop Guide for Performing SFVP Reviews; Revision 4.0; dated January 13, 2003
--	Desktop Guide for SFVP Review Oversight Panel; Revision 1; dated January 13, 2003
NG-NA-00305	Operating Experience Assessment Program; Revision 02; dated February 19, 1999
--	Operating Experience Reference Guide; Revision 3; dated July 31, 2002
NOBP-LP-2006	Collective Significance Review; dated January 24, 2003

System Specific Safety Function Validation Reports

--	System Specific Safety Function Validation Report for HVAC for ECCS Pump Rooms; dated January 23, 2003
--	System Specific Safety Function Validation Report for Decay Heat Removal/Low Pressure Injection System; dated January 24, 2003
--	System Specific Safety Function Validation Report for Steam Generator System; dated January 23, 2003
--	System Specific Safety Function Validation Report for Main Steam System; dated January 24, 2003
--	System Specific Safety Function Validation Report for High Pressure Injection System; dated January 24, 2003

-- System Specific Safety Function Validation Report for Safety Features Actuation System; dated January 23, 2003

-- System Specific Safety Function Validation Report for 125/250 Vdc System; dated January 24, 2003

-- System Specific Safety Function Validation Report for 480 Vac System; dated January 24, 2003

-- System Specific Safety Function Validation Report for Steam and Feedwater Rupture Control System; dated January 24, 2003

-- System Specific Safety Function Validation Report for 4160 Vac System; dated January 24, 2003.

<u>System Health Readiness Review Reports</u>

-- System Health Readiness Review - Decay Heat Removal/Low Pressure Injection; Revision 00; dated December 19, 2002

-- System Health Readiness Review - Station Blackout Diesel Generator (SBODG) and SBODG Heating, Ventilation, and Air Conditioning (HVAC) System; Revision 00; dated December 3, 2002

-- System Health Readiness Review - Containment System; Revision 00; dated December 30, 2002

<u>Latent Issues Review Reports</u>

-- Latent Issues Review - Auxiliary Feedwater System; Revision 00; dated November 21, 2002

-- Latent Issues Review - Emergency Diesel Generator (EDG) System; Revision 00; dated November xx, 2002

-- Latent Issues Review - Component Cooling Water System; Revision 00; dated November xx, 2002

-- Latent Issues Review - Service Water System; Revision 00; dated December 7, 2002

-- Latent Issues Review - Reactor Coolant System; Revision 00; dated December 10, 2002

Collective Significance Review Reports

--	HELB Program Collective Significance Review; dated March 4, 2003
--	Seismic Program Collective Significance Review; dated March 18, 2003
--	Environmental Qualification Collective Significance Review; dated March 19, 2003
--	Appendix R - Safe Shutdown Analysis Collective Significance Review; dated April 1, 2003
--	Station Flooding Program Collective Significance Review; dated March 19, 2003

Condition Reports Generated from Inspection

03-04847	NRC Concern with Resolution of CR 02-03351; dated June 19, 2003
03-04892	PCR/NG-DB-0800; Production/Generation Risk Evaluation Form; dated June 21, 2003
03-04924	NRC Questions on Seismic Collective Significance Report (SH Assurance Inspection); dated June 23, 2003

Condition Reports (CR)

01-03056	Non-Conformance of AFW Pump #2 for MSLB Break Concurrent with LOOP; dated November 13, 2001
02-03401	OE14216 Residual Heat Removal Piping Movement Out of Specification; dated July 5, 2002
02-04680	LIR-EDG: Do not have Documentation to Assure Compliance with GE SIL 44 for HFAS; dated August 21, 2002
02-05512	SHRR Testing Review CTMT System, Extension of CTMT Boundary; dated September 5, 2002
02-05479	LIR-CCW: Non-seismic Piping over Safety-related Components; dated September 10, 2002
02-06868	SHRR: Some Functions of the SBODG Governor Are Not Tested; dated September 27, 2002
02-06955	LIR-AFW-WO 00-2216 Soft Foot; dated September 28, 2002
02-07081	SHRR: During SBODG CR Review - Numerous Problems with Air Start System; dated October 1, 2002
02-07129	SHRR: Operating Experience not Documented; dated October 1, 2002

02-07236	LIR-AFW Steam Generator Accident Pressure vs AFW Pump Flow; dated October 2, 2002
02-07501	LIR-AFW Transition to Decay Heat Removal System; dated October 4, 2002
02-07782	Watertight Seals for Conduit Penetrations in Rooms 51 and 52; dated October 8, 2002
02-08925	Design Issues Identified During (SHRR) Review of DC System; dated October 31, 2002
02-09011	EQ Walkdowns: Potential Replacement of SOR Pressure Switch PSHRC2B4; dated November 4, 2002
02-09224	LIR Systems Latent Issues Review Collective Significance; dated November 11, 2002
02-09294	SHRR: Commitment O017446 for SBODG Testing Inaccurate; dated November 12, 2002
02-09224	LIR Systems Latent Issues Review Collective Significance; dated November 11, 2002
02-09298	SHRR: Potential Weakness in SBODG Dead Bus Test; dated November 12, 2002
02-10470	CATS Rollover - Flood and High Energy Line Break (HELB) Control; dated December 26, 2002
03-00534	SHRR Collective Significance of System Health Readiness Reviews; dated January 22, 2003
03-00656	Collective Significance of CRs Associated with the EQ Program; dated January 24, 2003
03-01517	Collective Significance of Flooding; dated February 24, 2003
03-05658	LIR Collective Significance Review CR Response Adequacy; dated July 15, 2003
03-05778	NRC Questions from the LIR/SHRR Collective Significance Reviews; dated July 18, 2003
03-06613	Seismic Collective Significance Review; dated August 15, 2003

<u>Drawings</u>

M033A	Piping & Instrument Diagram - High Pressure Injection; Revision 30
M033B	Piping & Instrument Diagram - Decay Heat Train 1; Revision 39

M033C	Piping & Instrument Diagram - Decay Heat Train 2; Revision 16
M042C	Piping & Instrument Diagram - Sampling System; Sheet 3; Revision 32
E-1 SH. 1	AC Electrical System One Line Diagram; Revision 21
E-3	4.16 kV Metering and Relaying One Line Diagram; Revision 30
E-7	125/250 VDC and Instrumentation AC One Line Diagram; Revision 21

Calculations

C-NSA-99-16.70	Fussell-Vesely Importance Values for Safety Related Systems; Revision 0; dated December 6, 2003
C-EE-002.01-010	DC Calc - Battery/Charger Size, Short Circuit, Voltage Drop; Revision 28

Other Documents

SD-042	System Description for Decay Heat Removal System, Revision 2
--	Davis-Besse Safety Function Validation Project (200-044) Task Specific QA Plan; Revision 1; dated January 13, 2003
--	Meeting Report - December 13, 2002 Safety Function Validation Project Oversight Panel; dated December 31, 2002
--	Meeting Report - December 27, 2002 Safety Function Validation Project Oversight Panel; dated December 31, 2002
--	Meeting Report - January 3, 2003 Safety Function Validation Project Oversight Panel; dated January 6, 2003
--	Discovery Phase Collective Significance Latent Issues Reviews and Associated Inspections; dated December 10, 2002
MPR-2487	Davis-Besse Nuclear Power Station Safety Function Validation Project; Revision 0; dated January 24, 2003
--	Potential Safety Consequences of Nonconforming Conditions at Davis-Besse Phase I: Identification of Key Safety Significant Issues/Questions; Revision 0; dated December 27, 2002
--	Discovery Phase Collective Significance System Health Readiness Reviews; dated February 12, 2003
--	USAR, Section 6.3; Emergency Core Cooling System; Revision 22

-- NUMARC 87-00; Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors

-- IEEE Std 450-1987; IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations; dated 1987

-- IEEE Std 485-1978; IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations; dated 1978

-- Probabilistic Safety Assessment for the Davis-Besse Nuclear Power Station - Summary Report; dated May 2002

-- Resolution of System Health Assurance Plan Design Issues; Revision 01; dated January 14, 2003

– Davis-Besse System health Assurance discovery Action Plan; Revision 2; dated October 15, 2002

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CCW	Component Cooling Water
CDF	Core Damage Frequency
CRDM	Control Rod Drive Mechanism
CR	Condition Report
CSR	Collective Significance Review
DBVP	Design Basis Validation Project
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ELMS	Electrical Load Management System
EQ	Environmental Qualification
ETAP	Electrical Transient Analysis Program
HELB	High Energy Line Break
HPI	High Pressure Injection
HVAC	Heating, Ventilation, and Air Conditioning
LIR	Latent Issues Review
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
MS	Main Steam
NRC	Nuclear Regulatory Commission
PDR	Public Document Room
RCS	Reactor Coolant System
SBODG	Station Blackout Diesel Generator
SFAS	Safety Features Actuation System
SFRCS	Steam-Feed Rupture Control System
SFVP	Safety Function Validation Project
SG	Steam Generator
SHA	System Health Assurance
SHRR	System Health Readiness Review
URI	Unresolved Item
USAR	Updated Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current