U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/97004
Licensee:	Toledo Edison Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5503 N. State Route 2 Oak Harbor, OH 43449
Dates:	March 3 - April 14, 1997
Inspectors:	S. Stasek, Senior Resident Inspector K. Zellers, Resident Inspector
Approved by:	John M. Jacobson, Chief Reactor Projects Branch 4

EXECUTIVE SUMMARY

Davis-Besse Nuclear Power Station
NRC Inspection Report No. 50-346/97004

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a six-week period of resident inspection.

Operations

- Good operator awareness of equipment and system status was noted. Shift turnovers and shift briefs were well conducted and effectively communicated important operational information (Section O1.1).

- Operator usage of and adherence to administrative, operating, and surveillance procedures was good (Section O1.1).

- Engineered safety features and important-to-safety systems were verified lined up in accordance with plant drawings and the updated safety analysis report (Section O2.1).

Maintenance

- Material condition of important standby equipment was excellent overall (Section O2.1).

- Maintenance personnel performed observed work activities in accordance with maintenance work order (MWO) work instructions and approved maintenance procedures in all cases (Section M1.1).

- Surveillance testing activities observed during the inspection period provided adequate assurance that the tested equipment could perform as designed (Section M1.1).

- The procedure for foreign material exclusion did not provide sufficient guidance to workers in some cases (Section M3.1).

- An ongoing failure to perform technical specification required monthly testing of undervoltage relays that initiate essential bus loadshed and emergency diesel generator start was considered a violation of NRC requirements (Section M3.2).

Engineering

- Several issues relating to the installation of the shield building blowout panels were identified during the inspection. The inspectors questioned whether caulking used to seal the edges of the panels could adversely affect the pressure at which they were set to blow out. A number of blowout panel shear bolts were not installed

per the installation drawings. There was no procedure guidance for shear bolt installation as well (Section E2.1).

- Information supporting the acceptability of installing armor plates on several station fire doors was not readily retrievable. As such, the licensee could not easily determine that use of armor plating was acceptable and declared the subject doors inoperable and established compensatory fire watches while completing their review (Section E2.2).

- Plant engineering identified that relays in the control circuits for all three service water blowdown valves had experienced age related degradation, and subsequently, all were replaced. In addition, the inspectors identified that a design flaw in the same control circuitry could cause the blowdown valves to misoperate as well, resulting in a portion of service water flow to be redirected from the normal safety loads. Similar relays and control logic were also used in the design and installation of other safety related valves. Extent-of-condition reviews were ongoing at the end of the inspection period (Section E8.6).

Plant Support

- NRC observed station activities were conducted in accordance with radiation protection (RP) program requirements (Section R1).

- Radiation, high radiation, and contaminated area controls and postings were in conformance with RP procedures (Section R1).

- An emergency plan drill was observed on April 2. The inspectors noted that drill objectives appeared to have been met, good participation by station personnel was evident, and the drill scenario was well controlled and realistically implemented (Section P1).

Report Details

Summary of Plant Status

The unit operated at nominally full power throughout the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors observed control room operators during their conduct of shift activities, walked down control room panels, reviewed logs, and held discussions with operations personnel during the inspection period. Good operator awareness of equipment and system status was noted. Shift turnovers and shift briefs were well conducted and effectively communicated important operational information. Operator usage of and adherence to administrative, operating, and surveillance procedures were good.

O2 Operational Status of Facilities and Equipment

O2.1 System Walkdowns (71707)

The inspectors walked down the accessible portions of the following engineered safety features (ESF) and important-to-safety systems during the inspection period:

- high pressure injection system - trains 1 and 2
- low pressure injection system - trains 1 and 2
- containment spray system - trains 1 and 2
- emergency diesel generators #1 and #2
- station essential batteries
- hydrogen dilution system - trains 1 and 2
- auxiliary feedwater - train 2
- motor driven feedpump

No substantive concerns were identified as a result of the walkdowns. System lineups and major flowpaths were verified to be consistent with plant drawings and the updated safety analysis report (USAR). Equipment material condition was found to be excellent in all cases. Pump/motor fluid levels were within their normal bands. Only very minor oil and fluid leaks were noted on occasion. Local and remote controllers were properly positioned and attendant instrumentation appeared to be functioning correctly.

O8 Miscellaneous Operations Issues (92700) (92901)

O8.1 (Closed) LER 50-346/94-002-00: *Anticipatory Reactor Trip System Inoperable Due to Failure to Trip Main Feedpump Turbine.* With the plant at approximately 55% power, operators removed the #2 main feedpump (MFP) from service and isolated the pump to repair a minor steam leak on an associated vent line. Because the main feedpump turbine was not actually tripped when it was removed from service, the turbine governor hydraulic control oil remained at normal pressure; however, the anticipatory reactor trip system (ARTS) logic monitored control oil pressure as an input for ARTS initiation. With the MFP shutdown at the indicated power level, the ARTS system could not have responded as designed.

The safety consequence of this event was minimal in that the ARTS system was designed to anticipate a reactor high pressure condition upon a trip of the main turbine above 40% power and/or loss of normal feedwater flow. The reactor protection system remained operable during this timeframe and would have provided adequate protection to the core in response to a loss of normal feedwater or a main turbine trip above 40% power. No credit for ARTS was taken in the licensee's USAR accident analyses.

Following the event, additional operator training was provided to address proper MFP shutdown/trip actions, and applicable procedures were revised to clarify MFP shutdown requirements and reduced power operation utilizing one MFP. In addition, the licensee prepared and implemented plant modification 95-0011, to install test toggle switches in the ARTS input logic. This allowed operators to input MFP availability status to ARTS, independent of whether a MFP was actually tripped.

O8.2 (Closed) Unresolved Item (50-346/96005-01(DRP)): Shift manager proficiency watchstanding requirements were vague. The licensee subsequently upgraded the guidance for shift manager proficiency watches, preliminarily in the form of a memorandum from the operations manager to all shift managers. The guidelines were in process of being translated to the shift manager qualification journals at the end of the inspection period.

O8.3 (Closed) Inspection Followup Item (50-346/96005-02(DRP)): Operations shift work schedules not consistent with technical specifications (TS). TS 6.2.3 was subsequently revised to better reflect current shift work schedules.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance and Surveillance Activities (61726) (62707)

The following maintenance and surveillance testing activities were observed/reviewed during the inspection period:

•	MWO 7-97-0235-02	Service Water Valve SW 1381 Troubleshooting
•	MWO 3-97-4402-01	59% Undervoltage Device Testing
•	MWO 3-97-1702-01	AFW #2 Turbine Governor Preventive Maintenance
•	MWO 3-97-0721-02	EDG #1 Quarterly Engine Checks
•	MWO 3-96-5321-02	ECAD Testing of AFW Train 2 Electrical Cables
•	MWO 1-96-0905-00	AFW #2 Pump Bearing Drainline Leak Repair
•	DB-ME-03040	SFAS Sequencer C1 Bus Undervoltage Relay Response Time Test
•	DB-MI-05254	Nuclear Instrumentation NI-05 (RPS Ch 2) Power Range Adjustment
•	DB-SP-03136	Decay Heat Pump 1 Quarterly Pump and Valve Test
•	DB-SS-03042	Control Room Emergency Ventilation System Train 2 Monthly Test

Maintenance activities observed during the inspection period were conducted in a controlled, coordinated manner. Pre-evolution briefs were thorough and performed in conformance with station requirements. Maintenance craft were sufficiently knowledgeable of equipment being worked on, and performed work in accordance with maintenance work order (MWO) instructions and maintenance procedures. In-field troubleshooting activities were well controlled and adequately supervised.

Surveillance testing activities observed during the inspection period provided adequate assurance that the tested equipment could perform as designed. The inspectors independently verified that the equipment functioned (under test conditions) per USAR descriptions. However, several undervoltage relays were found not being tested within the frequency required by technical specifications (reference Section M3.2).

M3 Maintenance Procedures and Documentation

M3.1 Foreign Material Exclusion Controls

a. Inspection Scope (62707)

The inspectors observed portions of an auxiliary feedwater (AFW) train 2 maintenance outage conducted on March 18.

b. Observations and Findings

The inspectors observed that the in-field maintenance activities were conducted in a controlled manner. Good coordination and communication were noted. In particular, the maintenance craft utilized appropriate foreign material exclusion (FME) controls during work on a drain line leak in the AFW pump inboard bearing oil system.

However, the inspectors noted the MWO (1-96-0905-00) work package for the bearing drain line leak did not provide for any FME controls during the job. Followup discussions with maintenance personnel revealed that the maintenance planner had prepared the MWO package in accordance with all required administrative controls including procedure DB-DP-00005, "Foreign Material Exclusion." The planner had determined that a "Foreign Material Exclusion Requirements" form did not need to be included in the MWO package because DB-DP-00005 allowed the work to be classified as not requiring FME controls. The inspectors were concerned that the procedure did not appear to provide sufficient guidance to the maintenance craft to preclude possible introduction of foreign material into the AFW bearing oil system. The maintenance crew working the job also had recognized that omission of FME controls from the MWO package was inappropriate and took additional actions to ensure FME was addressed during the work.

In addition, DB-DP-00005 was unclear in several areas, including the cleanliness classifications of important systems. For instance, AFW was categorized as a Class C system. However, the procedure did not indicate whether this included AFW turbine steam supply/exhaust, AFW condensate, as well as bearing oil.

The licensee agreed that additional review of station FME procedural controls should be conducted. At the conclusion of the inspection period, the licensee was evaluating additional areas and equipment for possible inclusion into DB-DP-00005. The procedure itself was also being reviewed in an attempt to better clarify pre-existing requirements. This matter is considered an **inspection followup item** pending resolution of this matter and subsequent inspector followup review **(50-346/97004-01(DRP))**.

M3.2 Essential Bus Loadshed/Emergency Diesel Generator Start Relays Not Tested in Accordance With Technical Specifications

a. Inspection Scope (61726)

The inspectors observed the performance of MWO 3-97-4402-01, "59% Undervoltage Device Testing" and reviewed associated procedures on March 25, 1997.

b. Observations and Findings

The relay observed under test was designed to function in concert with other similar relays in a logic scheme to sense a low voltage condition (59% of normal) on its assigned class 1E bus, initiate a bus loadshed, and send a start signal to the associated emergency diesel generator (EDG). Monthly testing of these relays to ensure proper tripping at the specified voltage and time delay was required by Technical Specification (TS) 3.3.2.1, "Safety Features Actuation System", Tables 3.3-4 and 4.3-2. The inspectors identified that the licensee was not verifying the 59% undervoltage device allowable trip setpoints for any of the relays on a monthly schedule.

Specifically, TS Table 3.3-4, under "Sequence Logic Channels" item b, included a note that indicated that allowable trip values should be verified during channel functional tests. TS Table 4.3-2 required that channel functional tests be performed monthly. However, the licensee was testing the voltage trip setpoint of the subject relays about every 4-6 months and testing the relay time delays every 18 months.

Following discussions with plant personnel, potential condition adverse to quality report (PCAQR) 97-0412 was promptly generated, and additional licensee review determined that the 90% undervoltage device relays had also not been tested at the appropriate frequency. The operating shift declared all of the 59% and 90% relays inoperable. Associated action requirement 15 of TS Table 3.3-3 was then applied which consequently required both EDGs to also be declared inoperable. In lieu of taking the required actions associated with both EDGs being inoperable (which included initiation of a plant shutdown within two hours), TS 4.0.3 was invoked which permitted a 24-hour delay in carrying out the actions associated with two inoperable EDGs in order to complete the subject testing.

Within the ensuing 24 hours, the plant conducted time response and voltage testing on all applicable 59% and 90% relays with no deficiencies noted. This allowed both EDGs to be declared operable. Subsequently, a monthly testing schedule for the relays was established.

During their review of this matter, the inspectors noted that engineering was in process of implementing a program to evaluate instrumentation testing associated with the technical specifications per NRC Generic Letter (GL) 96-01, "Testing of Safety-Related Logic Circuits." Although the individual reviews had just begun on a pilot basis, a charter had been prepared which included a number of review criteria and a sequence to follow when performing the details of the review. It was unclear whether the pilot program would have identified the 59% relay test frequency problem. However, following NRC identification that the EDG relays had not been tested within the required frequency, the licensee revised the charter to specifically include applicable "lessons learned," and to incorporate additional detailed review criteria to address this area. As a result, it appeared that the current review program would be sufficient to identify similar problems if implemented as intended. The GL 96-01 related reviews were anticipated to be completed before the summer of 1998.

c. Conclusions

The failure to perform setpoint verifications during monthly testing of the 59% undervoltage relays is considered a **violation** of TSs 3.3.2.1 and 4.3.2.1.1 and associated TS tables **(50-346/97004-02(DRP))**.

When tested, all relays met their required acceptance criteria, indicating that they would have responded as designed if called upon.

The engineering review program to evaluate the testing of safety related circuits in response to GL 96-01 appeared adequate to identify possible similar problems.

M8 Miscellaneous Maintenance Issues (92700) (92902)

M8.1 (Closed) LER 50-346/94-004-00: *Containment Hydrogen Purge Inlet Screen Not Installed.* This matter involved licensee identification during a refuel outage containment walkdown that the containment hydrogen purge (CHP) system inlet screen was not installed. The function of the screen was to prevent debris and foreign material from entering the inlet piping to the CHP system and impacting on operation of the containment isolation valves (CIVs), during and immediately following a loss of coolant accident (LOCA). During normal operation, the absence of the screen was not of concern in that no debris was postulated to be generated. Debris could only enter the system and affect the functioning of the CIVs during times when the valves were opened in conjunction with elevated containment pressures. However, the valves were only opened during either normal power operation or to control containment hydrogen concentration several days following a LOCA. As such, the potential for debris entry into the line was minimal.

The inspectors subsequently verified that the screen was reinstalled.

M8.2 (Closed) LER 50-346/94-005-00: *RPS Channel 4 Response Times Exceeded.* With the plant shut down and in mode 5, station testing activities identified that response times for reactor protection system (RPS) channel 4 functional units associated with reactor coolant (RC) low pressure and RC high pressure exceeded technical specification limits. The measured response times exceeded allowable limits by 0.125 seconds and 0.165 seconds for RC high pressure and RC low pressure respectably.

The licensee subsequently determined the cause for the increased response times was due to inadvertent installation of an incorrect buffer amplifier module in the RPS channel 4 logic cabinet. Safety consequences were determined to be minimal in that RPS channel 4 logic would still function to trip the unit at the appropriate setpoints. In addition, the other three RPS channels were verified to trip at both the appropriate setpoints and within the necessary response times. The RPS logic was configured to trip the unit upon tripping of any two out of four channels.

The incorrect buffer amplifier was replaced during the outage and RPS channel 4 was thereafter tested to ensure appropriate response timing.

M8.3 (Closed) Unresolved Item (50-346/94005-02(DRP)): Pilot operated relief valve (PORV) inadvertently opened during maintenance activities. During replacement of a reactor trip module (RTM) in RPS channel 1, the PORV inadvertently opened for approximately five seconds. RCS pressure decreased about 60 psi before the PORV reclosed. Troubleshooting determined that an electrical spike had occurred in the RTM causing a momentary erroneous high RCS pressure signal to the PORV.

Subsequently, the licensee determined that the RTM involved in this event was procured from the Sacramento Municipal Utility District (SMUD) along with several other spares. The subject RTM had been modified while at SMUD to an alternate configuration that caused the resultant problem. The licensee's receipt inspection had failed to identify the modified configuration. All similar SMUD RTMs were thereafter tested to ensure their proper configuration prior to returning them to stores.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Shield Building Blowout Panel Installation

a. Inspection Scope (37551) (71707)

The inspectors performed a walkdown of the shield building blowout panels and reviewed associated installation requirements on April 2.

b. Observations and Findings

During the walkdown, it was noted that caulking was used to seal the edges of the blowout panels. The inspectors subsequently questioned whether the adhesive properties of the caulking could affect the pressure at which the blowout panels would release. The shear bolts installed in the panels were designed to release at a very specific pressure range (0.65 to 0.67 psid). This difference amounted to approximately 100 lbs total pressure across the surface area of each panel. The inspectors were concerned that the addition of caulking could add greater than 100 lbs of additional holding strength.

The inspectors also reviewed the installation requirements associated with the blowout panel shear bolts. The inspectors determined that the guidance provided to personnel installing shear bolts was minimal. No procedural requirements were in place, nor were the installation instructions provided in associated drawings easily translated.

A subsequent walkdown of the blowout panels revealed that several shear bolts were not installed per the specified drawing requirements. Several bolts were installed without all necessary washers (no washer installed on the head side of the bolt). Five shear bolts were also found installed without use of locktite on the nuts to ensure that they remained in a "fingertight" condition. In addition, during inspection of the shear bolts, one shear bolt failed at much less than normal release pressure. At the conclusion of the inspection period, engineering was inprocess of evaluating the failure mechanism of the bolt, and further, reviewing the need to perform additional inspections of the blowout panels.

Pending completion of inspector review of the aforementioned issues, this matter is considered an **unresolved item (50-346/97004-03(DRP))**.

E2.2 Fire Door Armor Plates

a. Inspection Scope (71750)

The inspectors performed a walkdown of several fire doors during the inspection period. The inspectors reviewed door compliance with underwriter's laboratory (UL) certification requirements.

b. Observations and Findings

The inspectors noted that several fire doors installed in the plant utilized armor plates that appeared to be large kickplates. National Fire Protection Association (NFPA) standard 80 indicated that kickplates as large as 16 inches high could be utilized on fire doors without adversely impacting previous UL ratings. However, the standard indicated that larger kickplates should be tested and evaluated for acceptability. The licensee used armor plates/kickplates of a size that covered approximately 40% of door surface area in several applications in the plant. The inspectors requested the associated testing and/or UL approval documentation to ensure that the installed configurations were acceptable. At the end of the inspection period, the licensee determined that the information supporting the acceptability of installing armor plates on station fire doors was not readily retrievable. As such, the station could not easily determine that use of armor plating was acceptable and declared the subject doors inoperable and established compensatory fire watches while completing their review. Pending the retrieval of the subject information and subsequent inspector review, this matter is considered an **unresolved item (50-346/97004-04(DRP))**.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Violation (50-346/94016-02(DRS)): Failure to adequately control a design change resulting in use of non-qualified fittings on RCS attached piping. In response, the licensee performed an engineering evaluation of the installation and concluded that the interim use of the non-qualified parts was acceptable. A maintenance work order (MWO 7-94-1288-01) was initiated to replace the subject pipe fittings with those of a qualified type. This MWO was completed during the tenth refuel outage.

E8.2 (Closed) Inspection Followup Item (50-346/94016-03(DRS)): 10 CFR 50.59 safety reviews (safety evaluation screening process) appeared to be limited and narrowly focused in certain applications. The screening process did not ensure that all seven questions used to determine whether an unreviewed safety question existed, were evaluated. The licensee subsequently revised plant procedure NG-EN-00304, "Safety Review and Evaluation" including an attached safety review form (ED 7818-7) to better define and delineate screening criteria. In addition, training for applicable personnel was conducted to further define the screening requirements.

Emphasis on the function of structures, systems, and components, in addition to a literal application of the USAR, was reinforced.

E8.3 (Closed) Unresolved Item (50-346/95007-01(DRP)): Electrical equipment operating at greater than nameplate rating. Engineering subsequently performed a downstream load evaluation for busses YAU and YBU and determined the appropriate operating voltage ranges for each. The design criteria manual (DCCM-040) as well as the 120 VAC system description were revised to incorporate the modified output voltage limits. A check of rectifier YRF3 output voltage was added to the operators zone reading sheets with guidance incorporated to have electricians adjust the output voltage when needed to ensure conformance with nameplate rating. None of the subject equipment was determined to be inoperable.

The inspectors since have identified no additional electrical equipment operating outside of nameplate rating limits. As such, this matter is considered closed. The inspectors will continue to monitor electrical equipment operating parameters as part of the routine inspection program.

E8.4 (Closed) Unresolved Item (50-346/96003-06(DRP)): Appropriate compensatory measures not taken for inoperable radiant energy shields. This matter was subsequently reviewed and concluded with issuance of a violation in inspection report 50-346/96008.

E8.5 (Closed) Inspection Followup (50-346/96005-06(DRP)): Spent fuel reconstitution activities not described in the updated safety analysis report (USAR). Subsequently, USAR Section 9.6 was revised to include a description of those fuel reconstitution and fuel recaging activities anticipated to be performed in the spent fuel pool.

E8.6 (Open) Unresolved Item (50-346/97003-04(DRP)): Service water strainer blowdown valves failed to close on several occasions.

Relay Failure: No specific root cause for two similar failures of the service water strainer control circuit latch and trip coil to operate correctly had been determined by plant engineering. Therefore, age degradation was determined to be the general root cause. Consequently, the control circuit latch and trip relays for all three service water pumps were replaced with equivalent relays. During the engineering extent-of-condition review, it was found that these type relays were also installed in safety features actuation system components control circuits. Therefore, since safety related, risk significant equipment may also be affected by the aging phenomena, further inspector followup is warranted in this area.

Control Logic Issue: The inspectors determined that a separate failure of the service water strainer blowdown valve to properly operate, where the valve was found partially open instead of fully open as in the first two failures, was due to a design flaw. Upon review of the control logic drawings, the inspectors identified that the control logic design allowed for the service water strainer blowdown valves to deenergize at up to 25% open under certain conditions. When brought to the

attention of plant engineering, they agreed that the valve had functioned as designed. Consequently, at the end of the inspection period, plant engineering initiated a request to perform a modification to the blowdown valve control circuit to eliminate the design flaw.

Additionally, engineering determined that other control circuits in the plant may also have similar logic circuitry. Since this feature may be undesirable in other circuits as well, the inspectors will continue followup of this matter.

The inspectors investigated whether this design feature may have caused the service water system to have been inoperable in the past. The condition that was evaluated was for a design basis loss-of-coolant-accident, concurrent with a service water strainer blowdown valve 25% open, with ultimate heat sink temperatures at the TS limit of 85 degrees. Because of the relatively small diversion of service water flow (about 400 gpm), and because of design margins that were available in the containment air coolers, decay heat coolers, and other service water cooled equipment, the inspectors determined that the service water system would have still been able to perform its function as described in the updated safety analysis report.

E8.7 (Closed) Unresolved Item (50-346/97003-05(DRP)): Service water pump performance curve out-of-date. The inspectors found that the service water pump curve in the pump curves book did not match the curve attached to the service water pump quarterly surveillance test procedure. The curve in the pump curve book reflected pump performance for the originally installed pump. The curve in the surveillance test procedure reflected pump performance for a replacement pump that had been installed in 1995.

It was determined that the curve from the pump curve book was used as a baseline curve for the performance of calculation C-NSA-011.01-003, "Allowable Service Water Flow Diversion During Cold Weather". This calculation was performed in order to provide a basis for being able to bypass service water flow through a standby component cooling water heat exchanger during cool weather in order to assist in maintaining service water header pressure below the service water relief valve setpoint. The original curve was provided by the manufacturer and contained performance data over a much wider range than that of the replacement pump curve. The new pump curve data range was restricted by the limitations of the in-plant installation and was not as useful as the performance data supplied by the manufacturer.

The inspectors discussed this with the engineer who performed the calculation and reviewed the calculation itself to determine if the conclusions might have been adversely affected by use of the older curve. The inspectors determined that, due to the older pump curve performance data being more conservative than the newer pump curve, and because the old pump curve was purposely degraded to reflect worse case end-of-life and fouling characteristics, the use of the old pump curve instead of the new pump curve was conservative and had no adverse effect on the

conclusions of the calculation. Additionally, a revision to the calculation utilized the new pump curve and demonstrated that the original conclusions remained valid.

However, engineering personnel were planning as a corrective action, to add information to the pump curve book that indicated that the pump performance data that it contained reflected originally installed equipment, and not necessarily currently installed equipment. Because the plant had initiated appropriate corrective actions to address this issue, and because no safety consequence was identified, this matter is considered closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls (71707) (71750)

During the inspection period, the inspectors conducted frequent walkdowns of the radiological restricted area (RRA). Radiation, high radiation, and contaminated area controls and postings were verified to be in conformance with radiation protection (RP) procedures. In addition, the inspectors independently verified that actual area radiation levels were consistent with current radiological surveys and postings. An increase in the quality and quantity of radiological postings was noted this inspection period. Personnel involved in the performance of activities observed by the inspectors, conducted those activities in accordance with RP program requirements.

R2 Status of RP&C Facilities and Equipment (71707) (71750)

A sample of portable radiation survey instruments were inspected during the inspection period. The inspectors verified that they were functional and properly calibrated within required timeframes. Also, personnel contamination monitors (PCMs) located at the RRA exit point were verified to be functional and appropriately calibrated.

P1 Conduct of EP Activities (71750)

The inspectors observed portions of an emergency planning drill that was conducted on April 2, 1997. Drill objectives appeared to be met, good participation by station personnel was evident, and the scenario was well controlled and realistically implemented. Operations personnel manning the control room simulator appropriately implemented the emergency plan, effectively utilized emergency procedures and effectively controlled simulator equipment. Good utilization of the operations support center and technical support center (TSC) resources was made. Although course of action recommendations were made from the TSC, operations personnel thoroughly evaluated the recommendations to ensure their appropriateness. An effective post drill critique was conducted.

P2 Status of EP Facilities, Equipment, and Resources (71750)

The inspectors walked down the emergency control center (ECC), technical support center (TSC), and operations support center during the inspection period. All three emergency response facilities appeared to be well maintained and in a suitable state of readiness. Associated equipment appeared functional and appropriately staged to adequately support potential activation of the station emergency plan. Personnel access to the ECC and TSC was verified to be restricted as required by the licensee's program.

S2 Status of Security Facilities and Equipment (71750)

During the inspection period, the inspectors verified the integrity of protected area (PA) fence barriers, that PA isolation zones were appropriately marked and adequately free of foreign objects, and that PA lighting equipment and illumination levels were not noticeably degraded. Personnel processing facility ingress monitoring equipment was verified to be functional and appropriately maintained.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on April 14, 1997. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X3 Management Meeting Summary

On March 21, 1997, a meeting was conducted onsite to present the results of the NRC's most recent systematic assessment of licensee performance (SALP). NRC Region III management participating in the meeting included the Regional Administrator; Director, Division of Reactor Projects; Deputy Director, Division of Nuclear Materials Safety; and Chief, Reactor Projects Branch 4. The NRC Office of Nuclear Reactor Regulation was represented by the Project Manager for Davis-Besse. During the meeting, the ratings for each SALP functional area were discussed as well their bases and significant factors considered by the SALP board during their deliberations.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. K. Wood, Vice President, Nuclear
J. H. Lash, Plant Manager
R. E. Donnellon, Director, Engineering & Services
T. J. Myers, Director, Nuclear Assurance
L. W. Worley, Director, Quality Assurance
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D. R. Wuokko, Supervisor, Licensing
L. A. Bonker, Supervisor, Radiation Protection
G. W. Gillespie, Superintendent, Chemistry
T. J. Chambers, Shift Manager, Operations
D. L. Miller, Senior Engineer, Licensing
G. M. Wolf, Engineer, Licensing

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700:	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Plant Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-346/97004-01(DRP)	IFI	FME procedure unclear
50-346/97004-02(DRP)	VIO	Failure to perform monthly technical specification testing of undervoltage relays
50-346/97004-03(DRP)	URI	Blowout panel shear bolt installation
50-346/97004-04(DRP)	URI	Armor plate certification questioned

Closed

50-346/96005-01(DRP)	URI	Shift manager proficiency watchstanding requirements were vague
50-346/96005-02(DRP)	IFI	Operations shift work schedules not consistent with the technical specifications
50-346/94005-02(DRP)	URI	Pilot operated relief valve inadvertently opened during maintenance
50-346/94016-02(DRS)	VIO	Inadequate control of a design change
50-346/94016-03(DRS)	IFI	Safety reviews were limited and narrowly focused
50-346/95007-01(DRP)	URI	Electrical equipment operating at greater than nameplate rating
50-346/96003-06(DRP)	URI	Appropriate compensatory measures not taken for inoperable radiant energy shields
50-346/96005-06(DRP)	IFI	Spent fuel reconstitution not described in USAR
50-346/97003-05(DRP)	URI	Service water pump performance curve out-of-date
50-346/94-002-00	LER	Anticipatory reactor trip system inoperable due to failure to trip main feedpump turbine
50-346/94-004-00	LER	Containment Hydrogen Purge Inlet Screen Not Installed
50-346/94-005-00	LER	RPS Channel Four Response Times Exceeded

Discussed

50-346/97003-04(DRP)	URI	Service water strainer blowdown valve failures

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ARTS	Anticipatory Reactor Trip System
CFR	Code of Federal Regulations
CHP	Containment Hydrogen Purge
CIV	Containment Isolation Valve
ECAD	Electronic Characterization and Diagnostics
ECC	Emergency Control Center
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
FME	Foreign Material Exclusion
GL	Generic Letter
IFI	Inspection Followup Item
IR	Inspection Report
LER	Licensee Event Report
MFP	Main Feed Pump
MWO	Maintenance Work Order
NFPA	National Fire Protection Association
NI	Nuclear Instrumentation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PCAQR	Potential Condition Adverse to Quality Report
PCM	Personnel Contamination Monitor
PORV	Pilot Operated Relief Valve
psi	pounds per square inch
psid	pounds per square inch differential
RC	Reactor Coolant
RCS	Reactor Cooiant System
RP	Radiation Protection
RPS	Reactor Protection System
RRA	Radiological Restricted Area
RTM	Reactor Trip Module
SMUD	Sacramento Municipal Utility District
TS	Technical Specification
TSC	Technical Support Center
UL	Underwriter's Laboratory
VIO	Violation