March 22, 2004

RAM Item No. - LER-01 **Closed: Y**

Description of Issue - Review and Evaluate Main Steam Safety Valve Setpoints Greater than Allowable

Description of Resolution - Appropriate actions per Technical Specification (TS) 3.7.1.1 were taken, until each valve was adjusted and demonstrated proper operation within the allowable band. Licensee corrective action implementation is documented as closeout to Condition Report CR 02-0502.

Reference Material - Inspection Report 02-05, which is in ADAMS as Accession No. ml022060551.

RAM Item No. - LER-02 **Closed: Y**

Description of Issue - LER 50-346/2002-002 documented through-wall cracking in three control rod drive mechanism (CRDM) nozzles with pressure boundary leakage from Nozzle 3 and degradation of the reactor vessel head.

Description of Resolution - The finding described above is the result of a licensee performance deficiency. Specifically, the licensee failed to properly implement the boric acid corrosion control and corrective action programs, which allowed the reactor coolant system pressure boundary leakage to occur undetected for a prolonged period of time. The risk significance of this finding was evaluated by the NRC and determined to be of high safety significance (in the Red range) as documented by the NRC in inspection report 50-346/03-16 dated May 29, 2003.

An NRC Davis-Besse Oversight Panel was created in April 2002 to make sure that all corrective actions, required to ensure that Davis-Besse can operate safely, are taken before the plant is permitted to restart. The Panel was established under the Agency’s Manual Chapter 0350 and created a “restart checklist” categorizing 31 actions in seven major areas that must be completed before the NRC can make a restart decision. As of September, 2003, the Oversight Panel determined that the licensee had adequately completed 18 of those actions. The completed checklist items included Items Nos. 1.a & 1.b, associated with NRC review of the licensee root cause determinations as documented in NRC inspection reports No. 50-346/03-04 and 50-346/02-18. The NRC also reviewed licensee corrective actions associated with vessel head replacement as documented in 50-346/02-07. However, an outstanding Checklist item No. 2.a related to NRC reviews of the reactor vessel head replacement was open as of November, 2003, pending NRC review of the final acceptance testing of the replacement vessel head and the completion of hot control rod testing. These two specific remaining activities are adequately captured under CAL-04.

This LER was closed in Inspection Report 05000346/2003018.

Reference Material - NRC Inspection Report No. 50-346/03-18 (ADAMS Accession No. ml033080433).

March 22, 2004

RAM Item No. - LER-03 **Closed: Y**

Description of Issue - Review and Evaluate Fuel Movement in Spent Fuel Pool without required door LER.

Description of Resolution - This LER documents a situation where the emergency ventilation system was inoperable for approximately 37 minutes during a time when it was required to be operable. The full details of the event and associated closure are addressed in NRC Inspection Report No. 2002-005. Technical Specification 3.9.12 permits an emergency ventilation train servicing the spent fuel pool area to be considered operable when the containment equipment hatch is open and both doors of the containment personnel are open, provided at least one personnel air lock door is capable of being closed and a designated individual is available immediately outside the personnel airlock to close the door. During that 37 minutes, no designated individual was immediately available to close the personnel airlock door. Although this is a violation of TS 3.9.12, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the Enforcement Policy. Licensee corrective action implementation is documented as closeout to Condition Report CR 02-1199.

Reference Material - Inspection Report 02-05, which is in ADAMS as Accession No. ml022060551.

March 22, 2004

RAM Item No. - LER-04 **Closed: Y**

Description of Issue - Review and Evaluate containment isolation closure requirements for RCP Seal Injection Valves LER. As a result of this condition, during postulated accident conditions, a potential for uncontrolled radioactive leakage outside containment could be created. This condition has apparently existed since original plant construction, and is a violation of Technical Specification 3.6.3.1 for Modes 1-4. In addition, the valves were determined to be installed inconsistent with design assumptions. The causes of these conditions are less than adequate design interface communication and design control.

Description of Resolution - The licensee identified that the pressure regulating valve setpoint for the reactor coolant pump (RCP) seal injection was inadequate to ensure closure of the valves upon receipt of a containment isolation signal. The condition apparently existed since original plant construction. Downstream of these isolation valves are check valves that are designed to prevent flow out of the reactor coolant system, thereby isolating the flow path regardless of whether the RCP seal injection valves are closed. The test history of the check valves was determined to be highly reliable and had no test failures in the past 10 years. The regional SRA performed a Phase 3 assessment and determined that the issue had very low safety significance (green) due to the low initiating event frequency of an interfacing system loss of coolant accident (ISLOCA), 1E-7, coupled with the check valve's failure probability to prevent a potential ISLOCA if the RCP seal injection valve failed. The SRA also reviewed the licensee's risk assessment and determined that the calculation was conservative given the assumptions used. The licensee's analysis determined that the change in core damage frequency was in the 1E-8 range.

This LER was originally discussed in IR 50-346/02-10 and considered to be an Unresolved Item (URI) (URI 50-346/02-10-2), pending a formal evaluation of the risk imposed by this design issue.

LER 50-346/2002-004-00; Containment Isolation Closure Requirements for RCP Seal Injection Valves MU66A-D, was closed and URI 50-346/02-10-2 were closed in inspection report 2002-017. A licensee-identified violation associated with this issue is discussed in Section 4OA7 of the inspection report.

Licensee corrective action implementation is documented as closeout to Condition Report CR 02-02254, "RCP Seal Injection Air Operated Valves Will Not Perform Safety Function."

Reference Material - NRC Inspection Report No. 2002-017, which is in ADAMS as accession No. ml023430380.

March 22, 2004

RAM Item No. **-** LER-05 **Closed: Y**

Description of Issue **-** Review and Evaluate Containment Sump LER and supplement. See also Condition Reports 02-3859 & 02-5461

Description of Resolution **-** On July 3, 2003, a Significance and Enforcement Review Panel meeting was held regarding the significance of the failure to effectively implement corrective actions for design control deficiencies regarding containment coatings, uncontrolled fibrous material and other debris inside containment. This deficiency resulted in the inability of the emergency core cooling system sump to perform its safety function under certain accident scenarios due to clogging of the sump screen. The NRC staff determined that several combinations of factors found lead to core damage frequency increases in the 10^{-4} (Yellow) range.

NRC Inspection Report No. 50-346/03-15, issued on July 30, 2003, discussed this LER in detail and included issuance of a Preliminary Yellow Finding and Apparent Violation (AV) 50-346/03-015-05 for the licensee’s failure to effectively implement corrective actions for design control issues related to deficient containment coatings, uncontrolled fibrous material and other debris. The failure to effectively implement the corrective actions resulted in the inability of the emergency core cooling system sump to perform its safety function under certain accident scenarios due to clogging of the sump screen. The Apparent Violation was issued pending determination of the finding’s final safety significance.

FirstEnergy provided a written response dated August 29, 2003, acknowledging the performance deficiency. FirstEnergy did not contest the Finding. FirstEnergy's response provided no new information to change the NRC's preliminary conclusion. On October 7, 2003, the NRC issued the Yellow Final Significance Determination, which included a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to promptly identify and correct significant conditions adverse to quality involving the potential to clog the emergency core cooling and containment spray system sump with debris following a loss of coolant accident.

As corrective actions, FirstEnergy performed extensive modifications during the current outage on the sump. FirstEnergy replaced the previous emergency sump strainer with a much larger strainer. The unqualified coatings and other debris, including fibrous insulation remaining in containment, have been walked down, verified, and documented. Debris generation, transport, strainer head loss, and strainer integrity analyses were performed for the emergency sump to return the emergency sump to full qualification and operability. The NRC inspected FirstEnergy's new sump, which is documented in Inspection Report 50-346/03-06. The NRC concluded that the containment emergency sump design modification was consistent with the design and licensing basis requirements and based on field walkdowns the modification installation was adequately implemented consistent with the design. This issue is adequately resolved for restart and the LER will be closed in a future inspection report.

Reference Material **-** NRC Inspection Report Nos. 50-346/03-15 (ADAMS Accession Number ml032120360), 50-346/03-06 (ADAMS Accession Number ml031710897), and Generic Letter 98-04 response (ML033370836).

RAM Item No. - LER-06 **Closed: Y**

Description of Issue - Review and Evaluate EDG Missile Shield LER. See also Condition Report 02-5590 and URI-43.

Description of Resolution - This item was closed in Inspection Report 50-346/02-19 to an Unresolved Item. For the purposes of tracking, this item is closed, and the details will be discussed in the closure of URI-43 in this RAM, which was assigned to the Corrective Action Team Inspection (CATI) for resolution.

Reference Material - Inspection Report 50-346/02-19.

RAM Item No. - LER-07 **Closed: Y**

Description of Issue - Review and Evaluate Leakage from Incore Monitoring Instrumentation LER

Description of Resolution - The NRC evaluated the licensee's implementation of the NOP test and concluded that it provided reasonable assurance that there is no pressure boundary leakage of the RCS. Documentation of the NRC's review of the licensee's activities is in NRC Inspection Report No. 50-346/2003-023, which was issued on December 5, 2003.

Reference Material - NRC Inspection Report No. 50-346/2003-023 (ADAMS Accession No. ml033421074).

March 22, 2004

RAM Item No. - LER-08 **Closed: Y**

Description of Issue - This pertains to LER 05000346/2002-08-00, and -01, Containment Air Coolers Collective Significance of Degraded Conditions. Following unit shutdown in 2002, various degraded conditions were identified associated with the CACs, which were documented in several CRs. The issues were related to thermal performance degradation, and structural issues related to seismic adequacy, boric acid corrosion, and post accident thermal stress.

Description of Resolution - The LER documented thermal performance issues caused by cooling coil fouling conditions on the air (cooling fin) side, and water (inside tube) side which were identified. Additionally, foreign material (plywood) was found in the SW supply piping to CAC #2. In addition, two 10 CFR Part 21 reports were issued by the CAC control vendor and the motor vendor. The overall corrective action to resolve the physical degradation of the CAC units was the refurbishment of the units prior to plant restart. New CAC units were installed.

Programmatic improvements in reactor coolant system leakage control and boric acid corrosion management were implemented to prevent recurrence. The corrective actions were evaluated as acceptable for restart purposes.

During review of the LER, the team identified several concerns with the licensee’s evaluation. Because of the overall deficiencies in the licensee's evaluation, especially in regard to the thermal performance issue, the team was unable to agree with the licensee's conclusion that the CACs were operable during previous cycles.

The LER will remain open pending further review of past CAC degradation; specifically the extent of degradation and effect on the past safety function of the CACs. The additional reviews will provide information as to the ability of the CACs to provide cooling for the PORVs during feed and bleed operations during previous operating cycles.

Reference Material - NRC Inspection Report 50-346/03-10, Section 4OA6(b).b.1 (ADAMS Accession No. ml040680070) and LER 05000346/2002-008-00 and -01.

March 22, 2004

RAM Item No. - LER-09 **Closed: Y**

Description of Issue - On November 29, 2002, with the reactor defueled, it was discovered that the thermal sleeve connected to the 2-2 HPI/makeup nozzle had an axial crack. Inspection of the 2-1 HPI/makeup thermal sleeve also revealed a cracked thermal sleeve. No cracking was observed during the inspection of the remaining two HPI thermal sleeves.

Restart Checklist Item - 2.e

Description of Resolution - The remedial action was to replace the thermal sleeves. Inservice inspection procedures were developed to ensure enhanced inspection techniques would be used in the future to verify the integrity of the HPI/makeup thermal sleeves. The licensee stated that the visual inspections will include the use of high resolution video equipment and verification that the video equipment was applied in accordance with ASME Section XI, sub-article IWA 2210, “Visual Exam for VT-1 Examination.” The licensee stated that the frequency of inspection would be every other refueling outage. This LER is closed in NRC Inspection Report No. 50-346/03-10.

Reference Material - NRC Inspection Report 50-346/03-10 (ADAMS Accession No. ml040680070).

March 22, 2004

RAM Item No. - LER-10 **Closed: Y**

Description of Issue - Inability of Air-Operated Valves to Function During Design Basis Conditions

Description of Resolution - On January 30, 2002, with the reactor defueled, the licensee identified that several air operated valves (AOV) had negative operating margins and subsequently determined that eight valves were not capable of performing their safety functions for all required conditions as discussed below.

Valve MU3 is an air operated isolation valve which is normally open to allow letdown flow to pass from the letdown coolers to the purification demineralizers. Upon loss of instrument air, this valve is designed to fail closed. However, the licensee identified that upon loss of instrument air the spring force alone was not sufficient to close valve MU3 against maximum reactor coolant system differential pressure. The licensee implemented ECR 03-0111-00 to replace the valve actuator with a new larger piston actuator and nitrogen bottles which would ensure this valve shuts with design differential pressure.

Valve CC1495 is an air operated valve which is normally open to provide cooling water to non-essential components such as the spent fuel pool heat exchangers or reactor coolant pump seal return coolers. This valve is designed to close on a safety features actuation system(SFAS) Level 3 signal or a low level in the component cooling water (CCW) surge tank. However, the licensee identified that upon loss of instrument air, the air accumulator was undersized and would not ensure that the valve would fully close. The licensee implemented ECR 03-0136-00 to install a larger air accumulator associated with the actuator for valve CC1495.

Service water isolation valves SW1356, SW1357, and SW1358, are normally open to provide a flow path for SW to the containment air coolers (CACs). During normal and emergency operation two of the three CACs are in service and the remaining CAC will have its SW isolation valve closed to support containment isolation. However, with a loss of instrument air, the air accumulators for these valves did not have sufficient capacity to hold the valves shut for up to 30 days to support containment isolation. The licensee implemented ECR 02-0836 to install larger air accumulators.

Service water isolation valves SW1424, SW1429 and SW1434 serve as temperature control valves for by throttling SW flow through the CCW heat exchangers. During emergency operation, these valves go to their full open position upon receipt of an SFAS Level 2 signal to maximize SW flow through the CCW heat exchangers. These valves are required to fail open upon loss of instrument air. These valves have spring air cylinder actuators which require the presence of air to position the valve. Upon loss of instrument air, the licensee identified that these valves would not fully open. The licensee performed a dynamic differential pressure test (without instrument air) for valve SW 1434 and the valve only opened to 28 degrees from fully shut. The licensee initiated ECR 03-0299-00 to install an air accumulator for each valve, to provide safety related air during accident conditions.

The licensee performed a calculation of the as found conditions as described in LER 2003-001 to determine the increase in core damage frequency, core damage probability, large early release frequency and large early release probability due to the condition. Based upon the results of this calculation, the licensee determined that these valve conditions were considered to have minimal safety significance. The licensee's calculation and risk evaluation will be reviewed by a Senior Reactor Analyst (SRA) in Region III to confirm the licensee's conclusions.

The licensee implemented the corrective actions for each of the inoperable valves as stated above to restore these systems to an operable condition. In the original version of LER-03-001, the licensee had identified that the decay heat removal heat exchanger outlet valves CC1467 and CC1469 were not operable because of undersized operators. The licensee subsequently determined that these valves were operable based upon a revised calculation C-ME-016.04-035 “Component Level Review Calc for AOV CC1467/1469" which demonstrated that these valves had adequate operating margins with the existing accumulators. The inspectors reviewed calculation C-ME-016.04-035 to confirm that the licensee had used industry accepted methodologies to demonstrate sufficient operating margins (e.g. to account for uncertainties) existed for these valves to be considered operable. Because the licensee intended to apply the same approach to demonstrate operating margins for each of the modified AOVs, the inspectors did not identify any operability concerns for the modified AOVs.

Reference Material - The Mechanical Engineering Branch performed an in-office review in accordance with inspection procedure 71153 focused on evaluating the proposed licensee corrective actions documented in the LER and associated condition reports. Based upon this review, the Mechanical Engineering Branch will provide a report input to close this LER in an integrated resident Inspection Report after the SRA completes a review of the licensee's risk evaluation.

RAM Item No. - LER-11 **Closed: Y**

Description of Issue - Potential Degradation of High Pressure Injection Pumps Due to Debris in Emergency Sump Fluid Post Accident

Description of Resolution - LER 2003-002-00, and Supplement 1, dated 1/29/04 described corrective actions taken and presented the licensee's risk significance determination. Corrective actions included analysis, HPI pump modifications, qualification testing, in-plant testing, and removal of fibrous material from containment. NRR reviewed the overall approach to the modification of the high pressure injection pumps and concluded that the modification was acceptable and provided reasonable assurance that the HPI pumps will perform their required functions when called upon (TIA 2003-04, dated 02/11/04). Results of NRC's final significance determination will be documented in Inspection Report 05000346/2004005.

Reference Material - LER 2003-002-00; LER 2003-002-01, dated 1/29/2004; Task Interface Agreement 2003-04; Inspection Report 50-346/04-06.

March 22, 2004

RAM Item No. - LER-12 and URI-25 **Closed: Y**

Description of Issue - Requests for Issues: During the SSDI inspection in 2002, URI-25, Some Small Break Loss of Coolant Accident Sizes Not Analyzed, was identified. Specifically, it addressed concerns with the HPI pump minimum flow and deadhead (lack of flow) conditions (URI-25 and LER-12).

Description of Resolution - Following the questioning during the 2002 NRC SSDI inspection of a potential deadhead condition of the HPI pumps and the adequacy of thermal protection (minimum flow) for the pumps, the licensee performed a study, 86-5022260-00, to determine whether HPI pump operability during post-LOCA sump recirculation could be assured for all break sizes and transient scenarios.

This study identified a range of small break sizes from 0.00206 ft^2 (leak-to-LOCA transition area) to 0.0045 ft^2, which would result in RCS re-pressurization cycles that could continue following HPI pump realignment to the containment emergency sump and closure of the minimum flow recirculation valves. The study concluded that for this newly analyzed range of break sizes, past operability of the HPI pumps was a concern. This was because the re-pressurization cycles would result in a higher containment pressure than the shut-off head of the HPI pumps, resulting in pump dead heading (no flow), when HPI pump suction was from the sump.

Based on the results of the evaluation, several corrective actions were implemented. An additional minimum flow recirculation line was installed during RFO 13 for each HPI pump. For one pump, the line tapped off the previously existing minimum flow line and for the other a completely new recirculation line was installed. For both pumps, the new lines contained two isolation valves and a non-cavitating pressure breakdown orifice and connected to the LPI pump discharge upstream of its respective decay heat cooler for the corresponding safety train. The modification design specified a minimum 35 gpm flow rate (same as that specified for the original recirculation line) for pump protection when aligned to the emergency sump in "piggyback" operation with the DHR pumps. In this lineup, the decay heat coolers would provide cooling for the respective HPI Pumps without loss of sump inventory. Inspector concerns regarding the minimum 35 gpm flow rate were evaluated and resolved through URI-24 (see associated RAM closure form.)

Operator action would be required to open the valves on these additional recirculation lines prior to pump realignment from the BWST to the emergency sump. Because the postulated transient was a very slow developing scenario, the team determined that ample time would be available for operators to take this action. Additionally, the team confirmed that this action did not replace any existing automatic action. The licensee revised the emergency procedures to provide direction on establishing the HPI alternate minimum recirculation flowpath and provided training to the operators on its use.

These corrective actions were sufficient to resolve the concern addressed in the LER. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green).

Reference Material - NRC Inspection Report 05000346/2003010, Sections 4OA3(3)b.1 and 4OA3.(6)b.2 (ADAMS Accession No. ml040680070); URI 05000346/2002014-01o; and LER 05000346/2003-003-00 and -01.

RAM Item No. - LER-13 **Closed: Y**

Description of Issue - Temperature elements TERC3A5 and TERC3A6 [reactor coolant loop 2 hot leg wide range temperature elements], had not had their calibration verified as required by Technical Specification 3.3.3.6. This was due to these two RTDs not being included in the procedure that performs the Technical Specification required calibration and stability checks for reactor coolant system RTDs.

Description of Resolution - Upon discovery of this issue, the licensee verified the calibration of TERC3A5. Calibration verification was not possible for TERC3A6 due to the RTD being damaged during removal. Based on operational data obtained from all four reactor coolant hot leg wide range temperature instruments, the licensee believes that TERC3A6 would have passed its calibration verification. Since TERC3A6 was not calibrated in accordance with Technical Specification surveillance requirements, the licensee determined that the plant had operated with the non-calibrated instruments which represented a condition prohibited by Technical Specification and reportable under 10 CFR 50.73(a)(2)(i)(B)

The licensee performed an extent of condition and did not identify additional omission of RTD calibration requirements. On May 22, 2003, the licensee implemented procedure DB-SC-03159, "RTD Cross Calibration." This procedure replaced the original calibration procedure and will be used to determine the calibration accuracy and stability of the RCS narrow and wide range RTDs. This new procedure included TERC3A5 and TERC3A6 as part of the cross calibration process for reactor coolant system RTDs.

Although the failure to perform the required surveillance testing for TERC3A5 and TERC3A6 was a minor violation of Technical Specification 3.3.3.6, it constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The LER was reviewed by the inspectors and no findings of significance were identified.

Reference Material - NRC Inspection Report No. 50-346/2003-025, scheduled to be issued in January 2004.

March 22, 2004

RAM Item No. **-** LER-14 **Closed: Y**

Description of Issue **-** LER 2003-005, Revision 00 and 01, “Containment Gas Analyzer Inoperability Due To Isolation of Cooling Water

Description of Resolution **-** The licensee’s initial submittal of this LER discussed a condition where the component cooling water (CCW) isolation valves on the inlet and outlet to the heat exchangers located in each of the two Containment Gas Analyzers Systems (CGAS) were found stuck shut. This condition rendered the CGAS incapable of performing its design function. This issue was evaluated by the inspectors and documented in Inspection Report No. 50-346/03-17.

Supplement 01 to this LER, dated January 23, 2004, described the following two additional issues that were identified by the licensee during the extent of condition evaluation, that directly impacted the proper operation of the hydrogen gas analyzers:

- The instrument air supplied to the moisture trap drain check valve associated with the hydrogen analyzer’s heat exchanger was non-safety grade. Post accident, this air supply would not have been available and the drain valve would not have functioned. Additionally, the regulator which supplied the air to the drain valve was set at too low for the drain valve to operate as designed.

- The moisture trap’s potentially contaminated condensate, via the drain valve, would flow to a floor drain in a room not served by the emergency ventilation system. This constituted a potential containment bypass pathway.

The licensee addressed the first issue by eliminating the reliance of the drain check valve on instrument air by replacing the air operated drain valves with solenoid operated valves, powered by independent essential power sources. The second issue was corrected by routing the potentially contaminated condensate to an existing ECCS floor drain.

The LER is closed in Inspection Report No. 50-346/04-02.

Reference Material **-** NRC Inspection Report Nos. 50-346/03-17 (ADAMS Accession No. ml032721592), and 50-346/04-02.

March 22, 2004

RAM Item No. - LER-15 **Closed: Y**

Description of Issue - LER 50-346-2003-006;Potential Errors in Analysis of Block Walls Regarding HELB Differential Pressure and Seismic Events; dated July 21, 2003; Event Date May 21, 2003. The licensee, in the process of reviewing calculations associated with Davis Besse Auxiliary Building Structural Analysis, noted that a pressure caused by a high energy line break in Room 227(see attached diagram labeled figure 5 - Room 227 is a corridor/passageway on the 565' elevation of the Auxiliary Building) would also cause a pressure surge in the connected and right angle corridor labeled 241. That pressure surge, in combination with design seismic loads could cause failure of Wall 2257. Wall 2257 is a wall that is part of the partition that forms Room 240 which is the Boric Acid Addition Tank Room. Failure of Wall 2257 could adversely impact 2 Component Cooling Water Valves, Service Water piping to Containment Air Cooler 1, and the Boric Acid Addition System thus affecting systems required by Technical Specifications.

Description of Resolution - Under design change ECR 03-0297, the licensee initiated and implemented in August, 2003, a design that modified door 209 to open with a differential pressure low enough to preclude Wall 2257 failure. The door modification allows pressure in Room/Corridor 241 to be vented to Room 240 thus lowering the differential pressure across Wall 2257.

The inspectors have reviewed the operability evaluation and corrective actions, associated with CR 03-0397, that pertain to Wall 2257 and have physically verified that Door 209 has been modified to open in the event of a pressure spike in Corridor 241. From those reviews the inspectors conclude that the licensee can consider Wall 2257 operable and that remaining corrective actions are adequately scheduled for completion..

Reference Material - LER 50-346-2003-006; CR 03-02910; Root Cause Analysis Report for CR 03-02910 (Analysis of Masonry Walls) and CR 03-03937 (Masonry Wall Failure); Wall and Room diagram from Root Cause Analysis Report Masonry Wall.

RAM Item No. - LER-16 **Closed: Y**

Description of Issue - AC System Analysis Results Show Potential Loss of offsite Power Following Design Basis Accident.

Description of Resolution - Two issues were identified in the Licensee Event Report as follows:

- LER 2003-007-00 reported that if an SFAS level 4 actuation occurred when the plant was operating in Mode 1 at 100 percent power with degraded grid voltage (98.3 percent of nominal) and the electrical distribution systems aligned to a single startup transformer, the resultant reduction in voltage at the essential 4160 V buses would be of such magnitude and duration that the undervoltage relays would have tripped the supply breakers to the essential 4160 V buses. This would have resulted in an unanticipated Loss of Offsite Power.

- LER 2003-007-00 reported that if an SFAS level 4 actuation occurred when the plant was operating in Mode 1 at 100 percent power with degraded grid voltage (98.3 percent of nominal) and both startup transformers available, grid voltage would recover; however, voltages on the 480 V essential buses may not recover to the level necessary for the satisfactory operation of some essential loads.

For the first issue, the licensee discovered that the plant operated with only one startup transformer available several times in the past. Specifically, this condition existed on August 7, 2001, January 5, 2002, and August 13 and 14, 2001. Because of the previously described condition, neither offsite source was operable during these periods. Additionally, during the August 13 and 14 time period, since this condition was not known at the time, the EDGs were not tested within 8 hours, nor was the plant shut down to hot standby (Mode 3) conditions, as required by LCO 3.8.1 for a loss of both offsite sources. This constituted a violation of Technical Specifications.

For the second issue, the licensee, identified that under degraded voltage conditions, the following equipment may not be available: the control circuitry for the Main Feedwater Steam Generator Isolation Valves, one train of the Component Cooling Water Ventilation System, and one train of the Emergency Ventilation System. Since only one train of each of the ventilation systems may have been disabled, the remaining trains would be available to perform the intended function.

The inspector reviewed this LER and determined that the corrective actions associated with the issues reported appeared to be reasonable and adequate. The following corrective actions had already been completed:

- Change the tap settings on the 4160 V to 480 V essential substation transformers supplying power to essential buses E1 and F1 to increase the 480 V essential bus voltage during power operations.
- Change the tap settings on the 480 V to 240 V transformers feeding essential buses YE2 and YF2 to increase the 240 V essential bus voltage.
- Install interposing relays on the motor starters for the Main Feedwater Steam Generator Isolation motor-operated valves FW 601 and FW 612.
- Install shorting bars for selected hydramotor circuits.

- Revise the trip setpoint and Allowable Value for the 90% undervoltage essential bus feeder trip relays.
- Until the change to the Technical Specifications for the 90% undervoltage essential bus feeder trip relays is implemented, maintain administrative controls in accordance with NRC administrative Letter 98-10.

Additionally, the closure packages for these corrective actions associated with the LER were reviewed.

In their Integrated Report to Support Restart dated November 23, 2003, the licensee has committed to submit a license amendment request for revisions to the SFAS Technical Specification values by January 30, 2004.

Reference Material: - None.

RAM Item No. - LER-17 **Closed: Y**

Description of Issue - LER 2003-008-00 documented a condition where the licensee installed 5 relays in to the Safety Features Actuation System (SFAS) which were not rated for that application.

Description of Resolution - The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, having very low safety significance for failing to identify the appropriate corrective actions to address all significant causal factors related the configuration control aspects associated with the installation of unqualified relays in the SFAS system. This issue was considered NRC-identified because, even though two root cause evaluations were completed for two separate Significant Conditions Adverse to Quality (SCAQ) condition reports related to this issue (Technical Evaluation of Output Relay Issues for the Safety Features Actuation System (SFAS) (CR 03–02725), and Procurement of SFAS Relays (CR 03-03232)), neither evaluation addressed the configuration control aspects of how five generation 3 relays managed to be installed, prior to refueling outage 13, and not be identified as a causal factor in either of the two root cause evaluations. This issue has been entered into the licensee's corrective action program as CR 03-08556. This LER was closed in Inspection Report 05000346/2003018.

Reference Material - NRC Inspection Report No. 50-346/03-18 (ADAMS Accession No. ml033080433).

<u>RAM Item No.</u> - LER-18 **Closed: Y**

<u>Description of Issue</u> - Loss of offsite power due to grid disturbances.

<u>Description of Resolution</u> - As a result of the loss of power, the diesels started and restored station vital loads. However, as a result of stopping and automatic restarting of the Service Water Pumps, a pressure transient was experienced in the Service Water System which caused a gasket leak on a Component Cooling Water (CCW) Heat Exchanger and distortion of Containment Air Coolers (CAC) expansion bellows. The licensee attributed the distortion of the CAC Service Water piping expansion bellows to an inadequate hydrodynamic transient analysis which under-predicted the peak pressures that would occur for a loss of offsite power condition. The Service Water system leakage from the end bell of the CCW Heat Exchanger 3 was attributed to the inadequate thickness dimensions of the gasket material installed during the current extended outage. Corrective actions implemented included the replacement of the expansion bellows for all three CACs and the installation of the appropriately sized gasket on CCW Heat Exchanger 3. In addition, the licensee modified the valving to the CACs to significantly reduce the potential for water hammer in the service water supply to the CACs. These issues have been entered into the licensee's corrective action program as CRs 03-06590, 03-06597 and 03-06651.

<u>Reference Material</u> - Inspection Report No. 50-346/03-22.

<u>RAM Item No.</u> - LER-19 **Closed: Y**

<u>Description of Issue</u> - Potential Inoperability of Decay Heat/Low Pressure Injection System due to loss of valve disc pins.

<u>Description of Resolution</u> - The licensee determined that the loss of the taper pins was due to incorrect installation of the pins by the vendor coupled with vibration induced flow during valve testing and lack of adequate taper pin staking. The missing pins could not be located in the piping adjacent to the valve and are considered to be foreign material that may have been transported to the reactor vessel. The licensee performed an evaluation comparing this debris condition to Framatome ANP 51-501734-00, dated March 11, 2002, "Reactor Operation with Loose Parts at Davis-Besse." This evaluation compared the potential effects of specified taper pins to the potential effects of several steam generator plugs in circulation. The current condition was considered to be bounded by the Framatome evaluation relative to the effect on fuel fretting. The licensee evaluation concluded that there was low likelihood that the taper pins would cause fuel fretting. Framatome reviewed the licensee's evaluation and concurred with its conclusions. The licensee has taken corrective actions to properly install and stake the taper pins in DH14B. The licensee also ensured that the taper pins were correctly installed and staked in DH13A, DH13B, and DH14A (valves which utilized a similar disc taper pin arrangement). The licensee has taken additional corrective actions to ensure that maintenance procedures for the various butterfly valves installed at Davis-Besse have been revised to ensure that the taper pins or keys are staked or welded. Full details are addressed in NRC Inspection Report No. 50-346/2003-022.

<u>Reference Material</u> - NRC Inspection Report No. 50-346/03-22 (ADAMS Accession No. ml033570081).

March 22, 2004

RAM Item No. - LER-20 **Closed: Y**

Description of Issue - Inoperability of Containment Spray Pump (CSP) #1 due to the supply breaker tripping/opening upon a start or close signal. The breaker did close and then immediately tripped. Other similar breakers with similar loadings may be susceptible to similar tripping.

Description of Resolution -. The licensee determined that the cause of the trip was a malfunction in the ground fault function in the solid state trip device associated with the supply breaker for CSP 1. After determining that the ground fault trip was not required, the licensee processed a design change and removed the fault sensing circuit from containment spray pump 1 (Inspection Report No. 05000346/2003018). The licensee determined that the problem potentially involves all General Electric 480 V Model AK-25 metal clad circuit breakers with ground fault sensing and that the probability of a false ground fault trip increases with load. The licensee identified that CSP 2 is susceptible to potential false ground fault tripping and that 4 breakers that feed MCCs may have loadings similar to that seen by the CSP 1 breaker. The CSPs are the largest single 480 starting load with the plant. The licensee has accomplished a design change and removed the ground fault function of the CSP 2 supply breaker (CR 03-07794CA5). The licensee also reviewed the need for immediate action on the MCC breakers (CR03-10376). The licensee concluded that there was no history indicating an issue with ground fault false trips, that inrush current profiles seen by these MCC breakers was different from that seen by the CSP breakers, and that the MCC breakers are not susceptible to the type of trips experienced by the CSP breakers.

The licensee did not identify why the condition occurred at this time although one offsite expert suggested that trip device electronic drift might be responsible (Root Cause Analysis for CR 03-07794). The actions taken by the licensee resolve and correct the issue addressed in the LER. The LER will be dispositioned in Inspection Report 05000346/2003025 which covers the inspection period to December 27, 2003..

Reference Material - Licensee Event Report No. 50-346/2003-011, NRC Inspection Report No. 05000346/2003018 (ADAMS Accession No. ml033080433); Condition Report (CR) 03-07794; and CR 03-10376.

March 22, 2004

RAM Item No. - LER-21 **Closed: Y**

Description of Issue - Inoperability of Auxiliary Feedwater Pump Turbine 1 due to Governor Adjustment/Inadequate Response Time. On September 23, 2003, with the plant in mode 3, auxiliary feedwater pump 1 was being tested to verify that the pump would come to rated speed in the time required by the plant's existing analyses. The pump was originally declared operable but because of the non-inclusion of time required for instrument response in the Steam Feedwater Rupture Control System, the pump operability decision was reversed with subsequent testing revealing issues with the governor valve linkage and operability of associated steam traps. The subsequent testing also indicated variability in response time with test timing from the previous test.

Description of Resolution -. The licensee initially determined that the governor valve linkage misadjustments was the cause of slow turbine response. The need to revise the initial operability call was because of unclear wording in the test procedure. The governor valve linkage was removed and reassembled using an approved procedure, DB-MM-09098; AFPT Governor Maintenance; Revision 04 as reported in Inspection Report 05000346/2003018. Additionally the test procedures addressing AFW pump response time testing, DP-SP-03157 and DP-OP-3166, were revised to specifically include instrument response time in comparing results to acceptance criteria (CR03-07975CA10).

In the LER the licensee stated that the final conclusion was that the AFW pump did not meet acceptance criteria due to moisture in the steam line due to 2 performance degraded steam traps in the AFW pump steam supply piping. The licensee reworked the 2 traps plus others or inspected them (Root Cause Analysis Report for CR 2003-0795, Report Date November 18, 2003 and CR03-07975CA15) including the five traps mentioned in the LER. The licensee has also initiated a corrective action (CR03-07975CA19) to initiate PMs to address the periodic inspection and repair of AFW pump steam traps. The action has a due date of March 20, 2004.

The actions taken by the licensee should resolve the issue addressed in the LER and in Inspection Report 05000346/2003018. The LER will be dispositioned in Inspection Report 05000346/2003025 which covers the inspection period to December 27, 2003..

Reference Material - Licensee Event Report No. 50-346/2003-012 and NRC Inspection Report No. 50-346/2003-018 (ADAMS Accession No. ml033080433).

March 22, 2004

RAM Item No. - LER-22 **Closed: Y**

Description of Issue - Trip of Reactor Protection System During Plant Cooldown. Davis-Besse Operations Procedure DB-OP-06903, “Plant Shutdown and Cooldown,” Revision 11 provided an option to cooldown the reactor coolant system with the Group 1 Safety Control Rods fully withdrawn.

Description of Resolution - The performance deficiency associated with this event is the control room staff did not adequately monitor and control reactor coolant system pressure which resulted in an unexpected reactor trip on shutdown bypass high pressure. The licensee implemented several actions to address the deficiencies associated with this issue. These actions included the following:

- until completion of the initial analysis of the event, returned the plant to normal operating pressure and temperature with all control rods tripped;
- implemented an operational stand down until the initial investigation of the event could be completed;
- implemented their problem solving and decision process and developed a root cause analysis;
- modified the cooldown procedure to reflect results from the initial investigation;
- developed guidance for that cooldown that specified the pressure and temperature band specifically for each evolution until Mode 5 was attained;
- for the cooldown to Mode 5 for each control room shift crew, conducted just in time simulator training that covered just the activities that would occur on that shift; and
- chartered a team to review the collective significance of this trip and other recent events involving operations and to develop an operations improvement plan.

The licensee entered the event into its corrective action program as Condition Report 03-08374. The inspectors reviewed the outstanding corrective actions and determined the proposed actions and due dates assigned to these actions were appropriate.

Reference Material - Inspection Report Nos. 50-346/03-022 (ADAMS Accession No. ml033570081) and 50-346/04-02, scheduled to be issued in February 2004.

March 22, 2004

RAM Item No. **-** LER-23 **Closed: Y**

Date of Letter **-** LER 50-346-2003-014; Steam Feedwater Rupture Controls System Re-Energizes in a Blocked Condition; dated December 16, 2003; Event Date October 17, 2003

Description of Issue **-** The licensee has a Steam Feedwater Rupture Control System (SFRCS) that is designed to isolate feedwater to a ruptured steam generator and to provide feedwater to the non-faulted generator. In October, 2003, during work on the SFRCS which required de-energizing and energizing various portions of the system, the licensee identified that upon re-powering of logic channels they sometimes re-energized with the logic, that indicated a ruptured generator, in a blocked state. If a steam line break were to occur on steam generator 2, followed by a loss of offsite power, and if the susceptible logic circuits re-energized in a blocked condition, the SFRCS would not isolate feedwater to steam generator 2.

Description of Resolution **-** The initial event was described in IR 50-346/2003-022. The licensee developed and implemented a design change (ECP 03-0569-00) to correct the SFRCS logic to prevent blocking upon restoration of power. This modification fixed the physical discrepancy. Review of this modification was documented in IR 50-346/2003-025. The regulatory aspects of the LER and event will be documented in IR 50-346/2004-006. Based on review of the referenced documents, the inspectors conclude that the licensee can consider the SFRCS system operable and that open corrective actions have been adequately scheduled for completion.

Reference Material **-** IR 50-346/2003-022; IR 50-346/2003-025; Davis-Besse USAR Section 3.6.2.7.1.6; LER 50-346-2003-014; CR 03-08917.