March 22, 2004

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

Description of Issue - An apparent violation of the Davis-Besse technical specification associated with operation of the plant with pressure boundary leakage from through-wall cracks in the RCS.

Description of Resolution - The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues.

To evaluate the technical nature of the issue, the NRC conducted an inspection into the licensee's organizational management programs and reactor operations.

Operation with pressure boundary leakage beyond the technical specification action statement was a direct result of the licensee's failure to identify the control rod drive mechanism leakage, as noted in the findings from the NRC's Augmented Inspection Team Follow-up report (50-346/02-08(DRS)). The licensee's evaluation concluded that the specific programmatic issues, as identified in Licensee Event Report 2002-002-00, were an inadequate Boric Acid Corrosion Control (BACC) program and inadequate implementation of the Inservice Inspection (ISI) program.

Corrective action for the inadequate BACC program is discussed in inspection reports 50-346/02-11 and 50-346/03-09. Inadequate implementation of the ISI program was addressed through Self-Assessment 2002-081 and a Phase 2 program review by the Plant Review Board. Based on a review of these programs, the inspector determined that this issue was properly addressed by the licensee's corrective action program. The NRC's assessment of the effectiveness of those corrective actions are documented in the Corrective Action Team Inspection (CATI) report, No. 50-346/03-10. This item is considered closed for restart.

Reference Material - NRC Inspection Report Nos. 50-346/02-08 (ADAMS Accession No. ml022750524), 50-346/02-11 (ADAMS Accession No. ml031880844), 50-346/03-09 (ADAMS Accession No. ml031880844), and 50-346/03-10 (ADAMS Accession No. ml040680070); Condition Report (CR) 02-00891, "Control Rod Drive Nozzle Crack Indication," Licensee Event Report 2002-002, Root Cause Analysis Report - "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head," Davis-Besse Technical Specification, Limiting Condition for Operation for Reactor Coolant System Operational Leakage, paragraph 3.4.6.2, and Procedure DB-OP-01200, "Reactor Coolant System Leakage Management," Revision 5.

March 22, 2004

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

Description of Issue - Reactor Vessel Head Boric Acid Deposits.

Description of Resolution - The issue was the result of the licensee's failure to implement its corrective action and boric acid control programs. The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues. The inspectors examined the licensee's Root Cause Analysis Report on Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head. The causal factors for the issue were addressed on page 30 of the report and included:

- less than adequate safety focus
- less than adequate implementation of the corrective action program
- no safety analysis performed for the existing condition

Corrective actions were addressed globally by the licensee's Management and Human Performance Improvement Plan and the Program Compliance Plan. These were spelled out as corrective actions to CR 02-00891. Among the corrective actions were:

- Changes in corporate and plant senor management;

- Development of a management field presence/involvement plan to improve management oversight;

- Development of a management monitoring process to monitor and trend the performance of specific management oversight activities;

- Case study training for site personnel to include how the event happened, what barriers broke down, and what must be different in the future;

- Formal assessment of the safety conscious work environment at the plant based on criteria and attributes derived from NRC policy and guidance;

- Establish corporate-wide policy emphasizing the station's industrial and nuclear safety philosophy; and

- Realignment of management incentives to place more reward for safety and safe operation of the station.

Corrective actions for the failure to properly implement the corrective action program or to perform requisite safety analyses were specified under CR 02-00891. These directed a complete overhaul and reinstitution of the corrective action program. To ensure that safety analyses were performed as needed, corporate standards for analyses of safety issues were established and the use of a safety precedence sequence for root cause analyses was mandated. This was confirmed by the inspectors and considered adequate.

The root cause report also identified other, more discrete issues associated with this apparent violation. These included:

- addressing symptoms rather than causes
- less than adequate cause determinations
- less than adequate corrective actions

These were also addressed through corrective actions associated with CR 02-00891. Some of the corrective actions included a case study of this event with an emphasis on the need to find and address the causes of adverse conditions and the potential consequences of failure to do so, implementation of the Corrective Action Review Board to assess adequacy of actions and enforce higher standards for cause evaluations and corrective actions, mandating the use of formal root cause techniques coupled with independent reviews and self-assessments of cause evaluations, and improvements in effectiveness reviews with emphasis on verifying that causes have been properly addressed. These were confirmed by the inspectors.

The NRC's assessment of the licensee's effectiveness in implementing the revised corrective action program and the specific actions noted above is discussed in the CATI report (IR 03-10). This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002, and the Root Cause Analysis Report, "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; and NRC Inspection Report No. 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - Containment Air Cooler Boric Acid Deposits.

Description of Resolution - See text for closure of URI-02 above. The resolution for this item is identical. This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002, and the Root Cause Analysis Report, "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; and Inspection Report No. 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - Radiation Element Filters with Rust Deposits.

Description of Resolution - See text for closure of URI-02 above. The resolution for this item is identical. This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002, and the Root Cause Analysis Report, "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; and Inspection Report No. 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - Service Structure Modification Delay

Description of Resolution - This unresolved item addressed the licensee's repeated deferral of the modification to install multiple access ports in the service structure to permit cleaning and inspection of the reactor head. Modification 90-0012 was initiated in March 1990 to accomplish this but was deferred twice and then canceled in 1993. The modification was reinitiated in May 1994 as 94-0025 and subsequently deferred four times before the head degradation was identified in 2002.

The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues.

The licensee has resolved one portion of the URI through installation of the modification. The repeated deferral has been broadly addressed through the Management and Human Performance Improvement Plan and the Program Compliance Plan. In addition, CR 02-00891 resulted in a revision to the charter of the Plant Review Committee, which is the organization responsible for modification approval. The revision incorporated a requirement to include nuclear safety in the considerations when reviewing a plant modification.

The inspector concluded that the issues associated with this unresolved item had been properly addressed by the licensee's corrective action program. This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002; Root Cause Analysis Report "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; Request for Modification (RFM) 94-0025, "Install Service Structure Inspection Openings," Voided on August 31, 1999; EWR 01-0378-00, "Provide larger access holes to enable removal of boric acid," dated August 30, 2001; EWR 02-0138-00, "RV Service Structure Support Skirt Openings," dated April 11, 2002; and EWR 02-0217-00, "Replace Existing Reactor Vessel Head," dated June 4, 2002.

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

Description of Issue - Failure to follow the corrective action procedure and complete a prescribed corrective action for adverse trends in RCS unidentified leakage.

Description of Resolution - This unresolved item addressed the licensee's cancellation of a Mode 3 walkdown that was the proposed corrective action for an adverse trend in RCS unidentified leakage. Several months prior to the shutdown for the 2002 refueling outage the licensee had been examining increases in RCS leakage and as part of an extensive investigation, a walkdown of the containment while the plant was at normal operating temperature and pressure had been specified.

The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues.

The inspectors evaluated the licensees root cause report and corrective actions taken to address the issues identified in that report. The licensee's root cause analysis report identified the following causal factors for this item:

- less than adequate safety focus
- less than adequate implementation of the corrective action program
- less than adequate corrective actions

Corrective actions associated with the inadequate safety focus are addressed globally by the licensee's Management and Human Performance Improvement Plan and the Program Compliance Plan. These were spelled out as corrective actions to CR 02-00891. Among the corrective actions for these issues were:

- Changes in corporate and plant senor management;

- Development of a management field presence/involvement plan to improve management oversight;

- Development of a management monitoring process to monitor and trend the performance of specific management oversight activities;

- Case study training for site personnel to include how the event happened, what barriers broke down, and what must be different in the future;

- Formal assessment of the safety conscious work environment at the plant based on criteria and attributes derived from NRC policy and guidance;

- Establish corporate-wide policy emphasizing the station's industrial and nuclear safety philosophy; and

- Realignment of management incentives to place more reward for safety and safe operation of the station.

Corrective actions for the failure to properly implement the corrective action program or to perform requisite safety analyses were specified under CR 02-00891. These directed a complete overhaul and reinstitution of the corrective action program. To ensure that safety analyses are performed as needed, corporate standards for analyses of safety issues were established and the use of a safety precedence sequence for root cause analyses was mandated. This was confirmed by the inspectors and considered adequate.

The root cause report also identified other, more discrete issues associated with these apparent violations. These included:

- addressing symptoms rather than causes
- less than adequate cause determinations
- less than adequate corrective actions

These were also addressed through corrective actions associated with CR 02-00891. Some of the corrective actions included a case study of this event with an emphasis on the need to find and address the causes of adverse conditions and the potential consequences of failure to do so, implementation of the Corrective Action Review Board to assess adequacy of actions and enforce higher standards for cause evaluations and corrective actions, mandating the use of formal root cause techniques coupled with independent reviews and self-assessments of cause evaluations, and improvements in effectiveness reviews with emphasis on verifying that causes have been properly addressed. These were confirmed by the inspectors.

The inspectors concluded that this issue had been properly addressed by the licensee's corrective action program. This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002; Root Cause Analysis Report - "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; NOP-ER-3001, "Problem Solving and Decision Making Process," Revision 0; CR-01-2862, "Containment Inspection Plan Not Fully Implemented"; DB-OP-01200, "Reactor Coolant System Leakage Management," Revision 5; and NRC Inspection Report No. 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - The licensee's failure to have a boric acid corrosion control (BACC) program procedure appropriate to the circumstances.

Description of Resolution - The AIT follow-up inspection and the licensee's root cause report identified multiple deficiencies in the plant's BACC program procedure which contributed to the degradation of the reactor head.

The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues.

As part of the licensee's Program Compliance Plan, the BACC program procedure was completely revised and subjected to a phase 2 PRB review. The Program Compliance Plan, the PRB review, and the revised BACC program procedure were inspected and accepted by NRC inspectors; this inspection is documented in inspection reports 50-346/02-11 and 50/346/3-09.

The inspectors concluded that this issue has been properly addressed by the licensee's corrective action program. This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002; Root Cause Analysis Report, "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; Inspection Report 50-346/02-11 (ADAMS Accession No. ml031880844); and Inspection Report 50-346/03-09 (ADAMS Accession No. ml031880844).

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

Description of Issue - Failure to implement the boric acid corrosion control program procedure.

Description of Resolution - This unresolved item involved failure by the licensee engineering staff to follow a number of requirements of the boric acid corrosion control program procedure, most notably the requirement to remove all boric acid and examine the base metal underneath for signs of corrosion.

The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues.

The inspectors reviewed the sections of the licensee's root cause report which acknowledged these two issues, the section of the root cause report which outlined corrective actions, and the

corrective action specified under CR 02-00891. To correct the failure to follow the boric acid corrosion control program procedure, the licensee developed these specific actions:

- provide training to applicable personnel and mangers on the need to remove boric acid from components, to inspect for signs of corrosion, and to perform inspections for signs of boric acid in component internals; and

- reinforce standards and expectations for procedure compliance and the need for work practice rigor.

These were part of the licensee's global approach to the organizational effectiveness issue as part of the Management and Human Performance Improvement Plan and the Program Compliance Plan.

In the root cause, the licensee acknowledged that condition reports associated with the reactor head and other boric acid conditions were categorized as relatively low, which resulted in the use of simple cause analysis techniques. To address this, the licensee developed two corrective actions:

- Establish and ensure that criteria for categorization of the significance of repeat equipment failures are appropriate and used by station personnel. Criteria were to be sufficient to elevate repeat problems to higher levels, which require use of more robust analyses; and

- Review existing long-standing issues for possible elevation to significant condition status, thus engaging formal root cause evaluation techniques to obtain resolution of the issues

As part of the program compliance inspection and the corrective actions team inspection, both of these actions were verified to have been satisfactorily completed. This item is considered closed for restart.

Reference Material - CR 02-00891, "Control Rod Drive Nozzle Crack Indication," dated February 27, 2002; Root Cause Analysis Report, "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head"; Inspection Report No. 50-346/02-11 (ADAMS Accession No. ml031880844); and Inspection Report No. 50-346/03-09 (ADAMS Accession No. ml031880844), and Inspection Report No. 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - Failure to implement the corrective action program procedure.

Description of Resolution - This unresolved item involved failure by the licensee engineering staff to follow the guidance and examples for characterization of condition reports as significant, important, routine, or non-conditions adverse to quality and as a result repeatedly mischaracterized the conditions on the reactor head as routine.

The cause for this apparent violation remains under investigation. Any potential willful involvement in the apparent violation by an individual has been evaluated by NRC management and determined not to constitute an immediate safety issue. An NRR manager has been assigned to monitor the investigation and identify any potential safety issues.

The inspectors reviewed the sections of the licensee's root cause report which acknowledged these two issues, the section of the root cause report which outlined corrective actions, and the corrective action specified under CR 02-00891. To correct the failure to follow the boric acid corrosion control program procedure, the licensee developed these specific actions:

- provide training to applicable personnel and mangers on the need to remove boric acid from components, to inspect for signs of corrosion, and to perform inspections for signs of boric acid in component internals; and

- reinforce standards and expectations for procedure compliance and the need for work practice rigor.

These were part of the licensee's global approach to the organizational performance issue as part of the Management and Human Performance Improvement Plan and the Program Compliance Plan.

In the root cause, the licensee acknowledged that condition reports associated with the reactor head and other boric acid conditions were categorized as relatively low, which resulted in the use of superficial cause analysis techniques. To address this, the licensee developed two corrective actions:

- Establish and ensure that criteria for categorization of the significance of repeat equipment failures are appropriate and used by station personnel. Criteria were to be sufficient to elevate repeat problems to higher levels, which require use of more robust analyses; and

- Review existing long-standing issues for possible elevation to significant condition status, thus engaging formal root cause evaluation techniques to obtain resolution of the issues

As part of the program compliance inspection and the corrective actions team inspection, both of these actions were verified to have been satisfactorily completed. This item is considered closed for restart.

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Reference Material - CR 02-00891, “Control Rod Drive Nozzle Crack Indication,” dated February 27, 2002; Root Cause Analysis Report, “Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head”; Inspection Report No. 50-346/02-11 (ADAMS Accession No. ml031880844); and Inspection Report No. 50-346/03-09 (ADAMS Accession No. ml031880844), and Inspection Report No. 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - Completeness and accuracy of information. In the AIT report questions were raised regarding completeness and accuracy of documents either required by the USNRC to be maintained by the licensee or submitted to the USNRC.

Description of Resolution - NRC Inspection Report 03-19 reviewed the licensee's actions to resolve Restart Checklist Item No. 3.i., associated with the completeness and accuracy of required records and submittals to the NRC. The purpose of the inspection was for the NRC to determine whether reasonable confidence exists that important docketed information is complete and accurate in all material respects and that the licensee has taken appropriate corrective actions to ensure that future regulatory submittals are complete and accurate.

The inspection confirmed that the licensee has taken appropriate corrective actions to ensure that future regulatory submittals are complete and accurate in all material respects. The procedures for regulatory submittals have been revised to ensure that submittals are properly validated before issuance. Site personnel, including the site supervisory personnel, have been given training to ensure that they are cognizant of the requirements of 10 CFR 50.9 and the implications of not complying with those requirements. New supervisory training includes management responsibilities related to completeness and accuracy. New employee training includes the requirements of 10 CFR 50.9 as part of the orientation.

The inspection disclosed one particularly risk significant example regarding the licensee's response to Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant-Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment." As indicated in the Inspection Report, the licensee identified several corrective actions (CAs) as a result of this issue that have been completed which are:

- Update the response to Generic Letter 98-04 (Complete - CA 02-03-1718). The licensee's submitted a revised response to Generic Letter 98-04 on November 26, 2003 (ML033370836).

- Revise the UFSAR (Complete - CA 03-03-01718)

- Institute a Nuclear Safety-Related Protective Coatings Program (Complete - CA 02-02-03857)

- Institute an inventory of all non-Design Basis Accident (DBA) qualified coating materials (Complete - CA 04-02-02437)

- Removal and re-coating of Core Flood Tanks with DBA-qualified coating material (Complete - CA 03-02-03609)

- Removal and re-coating of Service Water piping with DBA-qualified coating material (Complete - CA 06-02-02108)

- Removal and re-coating of Reactor Vessel Head Service Structure with DBA-qualified coating material (Complete - CA 03-02-03609)

The completeness and accuracy inspection identified no widespread noncompliances of regulatory requirements or current programmatic concerns associated with the completeness and accuracy of submittals to the NRC. Based on the documents and corrective actions reviewed during this inspection and the results of previous NRC inspections of licensee activities under the Davis-Besse Return-to-Service Plan, the NRC has reasonable confidence that important docketed information is complete and accurate in all material respects and that future submittals will be complete and accurate.

The issue of the licensee providing complete and accurate information is closed for restart.

Reference Material - Inspection Report No. 50-346/02-08 and Inspection Report No. 50-346/03-19 and Inspection Report No. 50-346/02-03.

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

Description of Issue - Containment Isolation Closure Requirements for RCP Seal Injection Valves MU66AD. 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 - This issue is the same as for LER-04. See description of resolution for LER-04 for closure of this URI.

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

RAM Item No. - URI-12 **Closed: Y**

Description of Issue - Potential leakage at the reactor vessel in-core penetration tubes

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.

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

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

Description of Issue - Potential impact of corrosion on the ground function of electrical conduit in containment.

Description of Resolution - Condition Report 02-06788 described a condition where Boric acid corrosion of conduits in the containment could inhibit the flow of ground fault currents through the conduits (Conduits provide a supplementary grounding path for smaller motors). Further analysis, as documented in the Cause Analysis of CR 03-05239, determined that all conduits were acceptable as-is. The inspectors concurred with this conclusion.

Reference Material - Drawing No. E-1037P, Sheets 2 (Rev. 1), 3 (Rev. 1), 10 (Rev. 0) and 11 (Rev. 0); CR 02-06788; and CR 03-05239.

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

Description of Issue - During CAC motor replacement, the licensee identified splitting of the motor cable insulation as documented in CR 02-05459. The cable jacket and insulation to the three CAC motor high speed windings were found to be split at the ends which were normally covered by Raychem™ heat shrink sleeves. The damage was observed after the Raychem™ sleeves were removed for de-terminating the motors.

Description of Resolution - The NRC determined that the splitting was in fact a deep gash and the licensee subsequently determined the gash was inflicted by a contractor when removing the Raychem™ sleeves with a knife. To address this concern, the licensee initiated work orders to replace the section of the high speed cable of the three CAC motors between the motor and the penetrations with an equivalent cable. The work procedures were revised and the workers received training on the revised procedures. This item is resolved. The NRC determined this issue constituted a violation of 10 CFR Appendix B, Criterion V, (failure to properly remove Raychem™ splices during the CACs motor replacement), which has minor significance and is not subject to enforcement action.

Reference Material - NRC Inspection Report Nos. 50-346/02-14 (ADAMS Accession No. ml030630314) and 50-346/03-10 (ADAMS Accession No. ml040680070).

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

Description of Issue - Failure to include the environmental effects of a Decay Heat Removal (DHR) pump seal failure in its moderate energy line break analysis.

Description of Resolution - Following discovery, the licensee entered the issue into its corrective action program and performed the analysis. The NRC determined that the heat load caused by failure of the DHR pump seal (an additional 21,000 btu/hr) was subsequently included in calculation C-NSA-032.02-006 and that the issue was adequately resolved. A NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued.

Reference Material - NRC Inspection Report Nos. 50-346/02-14 (ADAMS Accession No. ml030630314) and 50-346/03-10 (ADAMS Accession No. ml040680070).

RAM Item No. - URI-16 and URI-34 **Closed: Y**

Description of Issue - URI-16, Lifting of Service Water Relief Valves, URI-34, Repetitive Failures of Service Water Relief Valves.

Description of Resolution - The issue dealt with a continuing operating condition where the relief valves on the tube (SW) side of the CCW heat exchangers would open during pump swap overs under low flow conditions such as winter operation with low heat loads. The frequent opening caused the valves to fail at an undesirable rate. The licensee had resolved the problem of inadvertent openings by changing the operating procedures. The inspectors verified that the valves were appropriately sized and set up correctly for the application. The reduction in inadvertent openings also resulted in a reduction of valve failures. The team concluded that the corrective actions implemented were reasonable to resolve this issue and there are no restart constraints. The team also concluded that there were no violations so the unresolved items are closed.

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

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RAM Item No. - URI-17 and URI 18 **Closed: Y**

Description of Issue - URI-17 concerned non-conservatisms in the analysis which analyzed the heat loads in the SW pump room and the ability of the ventilation system to maintain the pump room temperatures within a required operating range.

URI-18 dealt with the effects of a postulated auxiliary steam line break in the SW pump room and whether the licensee correctly translated the USAR commitments regarding the SW pump room environmental limits into analyses that demonstrated these limits would not be violated for design basis conditions.

Description of Resolution - Both of the items were examined together during the CATI. The heat load calculation was revised and issued as Revision 4 in early 2003. At the same time, another CR, was issued because the initial CR failed to do an extent of condition review to verify the adequacy of the SW ventilation system for all operating conditions. The extent of condition review was reported to have included a walkdown of the SW pump room and review of the revised SW ventilation calculation.

Upon review of the revised calculation in 2003, the NRC noted that the summer maximum analyzed temperature in the pump house did not include the heat load contribution of the diesel driven fire pump, which was one of the deficiencies noted in the earlier revision to the calculation. This deficiency was not addressed in the new revision to the calculation, either by including it or by providing a rationale for excluding the heat load. The NRC noted that the licensee had previously had to take actions to open the diesel generator room doors and provide alternate ventilation during the summer months. The new calculation also concluded that the penthouse louvers had to be modified (blocked) for winter operation. The NRC noted that past operability had been assured for winter operation by regularly recording pump room ambient temperature.

The NRC determined that past licensee compensatory actions (both during the summer and winter months) had prevented the equipment from being inoperable. An NCV of 10 CFR Part 50, Appendix B, Criterion III was issued in NRC Inspection Report 05000346/2003010 for URI-17.

The licensee performed a calculation assessing the environmental effects of a postulated auxiliary steam line break in the SW pump room. The calculation concluded that there were no adverse effects on the equipment in the room. The NRC also noted that the licensee had initiated engineering change request to remove the auxiliary steam line from the SW pump room. The licensee stated that this modification was an enhancement which was not required. There was no violation identified regarding RAM item URI-18.

Reference Material - NRC Inspection Report 05000346/2003010, Section 4OA3(3)b.7 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01e and 01f.

RAM Item No. - URI-19 **Closed: Y**

Description of Issue - On September 24, 2002, the licensee issued CR 02-06893 to document an increase from 95°F to 124°F in Rooms 105 and 115 temperature as a result of an increase of SW temperature. The CR identified the need to reevaluate cable ampacity as a result of the higher room temperature.

Description of Resolution - The team discussed the ampacity issue with the licensee, and determined there actually was not an ampacity concern. Therefore, this item is considered closed.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01g.

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

Description of Issue - The licensee failed to have provisions in place to protect the service water pump room from flooding.

Description of Resolution - During the SSDI in 2002, the NRC identified that no procedures were in place to isolate equipment open for maintenance in the SW pump room that could flood the room in the event of high lake water level. Therefore, the NRC questioned whether the SW system was adequately protected against flooding effects that could result from high lake water levels, from internal flooding, and from other threats to the system that could result from failure of non-seismically qualified equipment, as described in the USAR.

In response to this concern, the licensee determined that operator actions were necessary in order to ensure that the USAR statements were met. In order to ensure that the operator actions occurred, several changes to operating procedures were required. These procedural actions were taken, and this item is resolved.

A NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued.

Reference Material - NRC Inspection Report 05000346/2003010, Section 4OA3(3)b.9 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01h.

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RAM Item No. - URI-21 and NCV-9 **Closed: Y**

Description of Issue - URI-21 concerned the use of insufficiently supported uncertainty values in the calculation for the 90 percent Undervoltage Relays.

NCV-9 dealt with non-Conservative Relay Setpoint Calculation for the 59 Percent Undervoltage Relay

Description of Resolution - These two items were examined together during the CATI. The licensee performed additional analysis to assess the impact of using insufficiently supported uncertainty values.

The design remained adequate and there was no violation identified. URI-21 which was closed in NRC Inspection Report 05000346/2003010.

The NRC reviewed ETAP calculation C-EE-015.03-008, Revision 2. The calculation properly addressed the postulated inconsistencies and non-conservative assumptions in the uncertainty analysis. Therefore, the corrective actions to NCV 9 were evaluated as acceptable.

Reference Material - NRC Inspection Report 05000346/2003010, Sections 4OA3(2)b.7 and 4OA5(1)b.2.11 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01j and 01k.

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RAM Item No. - URI-22 and URI 23 **Closed: Y**

Description of Issue - URI-22, 05000346/2002014-01l, Inadequate Calculations for Control Room Operator Dose (GDC-19) and Offsite Dose (10 CFR Part 100) Related to High Pressure Injection Pump Minimum Flow Values, regarded concerns with the dose calculations for operators and the general public following a design basis accident. The licensee failed to translate the radiological consequences of leakage from engineered safety feature components outside containment into calculations of record for post-accident control room dose and offsite boundary dose.

URI-23, 05000346/2002014-01m, Oother GDC-19 and 10 CFR Part 100 Issues, is associated with correctly translating USAR commitments regarding calculations for GDC-19 and 10 CFR Part 100 requirements. During SSDI, the NRC determined that the USAR calculated offsite dose was based on an ECCS leakage rate of 1.6 gallons per hour (gph) while the allowable leakage rate was based on 40 gph.

Description of Resolution - Both items were examined together due to their similarity during CATI. The licensee performed a preliminary calculation in the cause analysis for CR 02-07701 to determine the increase in dose in the control room from the 500 gallons deposited in the Borated Water Storage Tank (BWST). The licensee then calculated that the total offsite dose was 236.22 rem. The total control room dose was similarly for a total of 20.366 rem.

As a result of these calculations, the licensee specified post-restart corrective actions to update the Bechtel calculation of record and the USAR to incorporate these doses. Because the corrective actions had not yet been completed, the licensee had not completed a screening or evaluation under 10 CFR 50.59. The team performed a limited evaluation of the acceptability of the increased dose under 10 CFR 50.59(c)(2)(iii), "Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated)." The team reviewed the guidance provided in Nuclear Energy Institute (NEI) standard 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, which NRC endorsed in Regulatory Guide 1.187. The team concluded that the licensee had an acceptable rationale for delaying issuance of the formal calculations until after restart.

The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance. Specifically, the licensee failed to translate the postulated radiological consequences of leakage from engineered safety feature components outside containment into calculations of record for post-accident control room dose and offsite boundary dose.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070), URI 05000346/2002014-01l, and URI 05000346/2002014-01m.

March 22, 2004

RAM Item No. - URI-24 **Closed: Y**

Description of Issue - Requests for Issues: During the SSDI inspection in 2002, the team identified an issue with the ability of the HPI pumps to perform as intended during extended operation on minimum flow. (URI-24).

Description of Resolution - The design requirements for the HPI system include the ability of the system to function at 35 gpm minimum flow. The licensee performed a six-hour test run on one of the pumps using the originally-installed minimum flow recirculation line and could not achieve flow rates below 53 gpm. On February 8, 2004, the licensee completed Operability Evaluation 04-004, Revision 1, which concluded that the HPI pumps were operable.

Based upon observed pump and system conditions during the test, the licensee concluded that the pump would remain operable at or near that flow. Due to the size of the installed orifice in the line, the inspectors concluded that it was reasonable that the pump would not experience flows much below that value and, in fact, the lowest obtained, recorded value noted for either pump during a review of surveillance tests conducted between June 2001 and December 2003 was 49 gpm. Also, based upon observed pump and system conditions during the six hour test run and feedback from the pump vendor, the licensee concluded the pumps would be able to run at minimum flow for extended periods of time during the designated mission time of 30 days post-accident. The inspectors agreed with the licensee's determination that the pumps would be able to perform their safety function. Further, the licensee planned several procedure changes to ensure actions would not be taken in the future that would reduce the minimum flow rate observed during the six hour test.

A NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued since the licensee had previously failed since initial plant startup to verify that the HPI pumps could operate under design basis minimum flow requirements. The inspectors have no restart concerns regarding this issue.

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

March 22, 2004

RAM Item No. - URI-25 and LER-12 **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. - URI-26 **Closed: Y**

Description of Issue - The licensee failed to perform an adequate SW flow analysis.

Description of Resolution - The licensee entered the issue into its corrective action program and performed the necessary calculations. The licencee initiated CR 03-03977 to revise the calculations. The team reviewed these calculations, evaluated the issue and identified several errors in the calculations that did not affect the design function of the system. The NRC staff concluded that there are no outstanding concerns for restart. NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01p.

RAM Item No. - URI-27 **Closed: Y**

Description of Issue - The licensee failed to perform an adequate SW thermal analysis.

Description of Resolution - The licensee entered the issue into its corrective action program and performed the necessary calculations. The licencee initiated CR 03-03977 to revise the calculations. The team reviewed these calculations, evaluated the issue and identified several errors in the calculations that did not affect the design function of the system. The NRC staff concluded that there are no outstanding concerns for restart. NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01q.

RAM Item No. - URI-28 **Closed: Y**

Description of Issue - The licensee failed to provide an analysis which addressed the service water valve single failure assumptions mentioned in the updated safety analysis report, specifically dealing with the ultimate heat sink's temperature and level. Specific combination included having design basis low ultimate heat sink levels and the system going into backwash while the system was responding to a design basis accident.

Description of Resolution - The licensee entered the issue in its corrective action program in Condition Report 03-06507. As an interim measure, the licensee implemented changes to operations procedures to close the affected service water valves. The licensee is also performing additional reviews and evaluations of the facility's conformance with design and licensing basis documents. These actions resolve any potential operability concerns regarding postulated single failures with maximum system temperatures and minimum heat sink level conditions.

This RAM item is closed with the stated actions being acceptable for plant restart. A NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01r.

RAM Item No. - URI-29 **Closed: Y**

Description of Issue - The licensee failed to perform a valid service water pump net positive suction head analysis, specifically the licensee's calculations determined that under a certain combination of design basis conditions pump net positive suction head (NPSH) was not achievable.

Description of Resolution - The licensee entered the issue into its corrective action program and performed the necessary calculations. The team reviewed these calculations and determined that there was one case where there was insufficient NPSH. The licensee initiated CR 03-03977 to revise the calculations. Following evaluation of CR 03-03977, the licensee concluded that the service water system is able to perform its safety-related function. The team agreed with the licensee's conclusions. The team also concluded that there were no related constraints for restart.

A NCV of 10 CFR Part 50, Appendix B, Criterion III, having very low safety significance (Green) was issued.

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

RAM Item No. - URI-30 **Closed: Y**

Description of Issue - This URI, 05000346/2002014-01t, Service Water Source Temperature Analysis for Auxiliary Feedwater, regarded the licensee failing to analyze the service water source with respect to its potentially higher temperature condition for various design basis events and the possible impact on the ability of the Auxiliary Feed Water (AFW) system to perform its safety function. Such effects could include reduced heat absorption capability for AFW injected into the SGs and inadequate cooling of AFW lubricating oil.

Description of Resolution - The licensee's evaluation concluded that temperature of AFW (seismic event with long term AFW supplied by SW) was lower than the design AFW temperature of 120°F as noted in the system description. In addition, the licensee determined that AFW equipment temperature limits were greater than 120°F. Therefore, the licensee concluded that there was no discrepant condition. The team agreed with this assessment and did not identify any violation. This item is resolved.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01t.

RAM Item No. - URI-31 **Closed: Y**

Description of Issue - The licensee failed to consider the worst-case grid voltages for the short circuit analyses performed in support of breaker coordination.

Description of Resolution - The licensee entered the issue into their corrective action program and performed new calculations to address the issue. The team reviewed this item and determined that calculation C-EE-015.03-003 was superseded with calculation C-EE-015.03-008. The new calculation did take into account the worst-case grid voltage conditions, and no other problems were identified. A violation of 10 CFR Appendix B, Criterion III, which has minor significance was identified.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-01u.

RAM Item No. - URI-32 **Closed: Y**

Description of Issue - During the 2002 SSDI, the NRC identified that the service water flow balance test procedure did not establish flows to the safety-related heat exchangers based on worst-case design basis conditions, such as degraded SW pumps, lowest UHS level, highest resistance SW system lineup, or system resistance degradation. Further, no analyses existed that established the test acceptance criteria for design basis conditions. This URI was written to document concerns with the flow balance testing for the SW system.

Description of Resolution - Following discovery, the licensee placed the issue in its corrective action program, evaluated it, and put procedures in place to address the issue. The licensee performed a service water flow balance test using revised procedures late in the outage. The results of the test were reviewed by the resident inspectors and the results documented in inspection report 50-346/03-25. The inspectors determined that the test was appropriately performed and the results met their design margin. The inspectors concluded that there were no constraints for restart.

A NCV of 10 CFR Part 50, Appendix B, Criterion XI, having very low safety significance (Green) was issued.

Reference Material - NRC Inspection Report Numbers 50-346/03-25 (ADAMS Accession No. ml040290768) and 50-346/03-10 (ADAMS Accession No. ml040680070).

RAM Item No. - URI-33 **Closed: Y**

Description of Issue - The licensee failed to identify a condition where the allowable degradation of the SW pumps did not match the design basis required flow rate for the SW pumps. In particular, the pump curve was allowed to degrade by 7 percent in accordance with IST acceptance criteria, without evaluating the required design basis flow requirement.

Description of Resolution - Vendor calculations 02-123 and 02-113 were performed to address all SW hydraulic issues. The allowable SW pump degradation was included in the new calculations. The team did not identify any violation and this item is closed.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-03a.

RAM Item No. - URI-34 and URI-16 **Closed: Y**

Description of Issue - URI-34, Repetitive Failures of Service Water Relief Valves, URI-16, Lifting of Service Water Relief Valves.

Description of Resolution - The issue dealt with a continuing operating condition where the relief valves on the tube (SW) side of the CCW heat exchangers would open during pump swap overs under low flow conditions such as winter operation with low heat loads. The frequent opening caused the valves to fail at an undesirable rate. The licensee had resolved the problem of inadvertent openings by changing the operating procedures. The inspectors verified that the valves were appropriately sized and set up correctly for the application. The reduction in inadvertent openings also resulted in a reduction of valve failures. The team concluded that the corrective actions implemented were reasonable to resolve this issue and there are no restart constraints. The team also concluded that there were no violations so the unresolved items are closed.

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

RAM Item No. - URI-35 **Closed: Y**

Description of Issue - This was a potential nonconservative temperature measurement performed by the licensee for ultimate heat sink temperatures.

Description of Resolution - The team determined that the licensee's procedures had been revised to incorporate the temperature instrument's uncertainty calculation results, and that the procedures required the plant staff to take appropriate actions should it appear that the ultimate heat sink temperature was being approached (such as measuring the temperature locally with sensitive measuring and test equipment.) Therefore, the team determined that no violation existed and this issue is closed.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-03c.

RAM Item No. - URI-36 **Closed: Y**

Description of Issue - Licensee failed in overestimating the nozzle flexibility by a factor of one thousand when analyzing the structural integrity of the connection in the SW system to the CACs.

Description of Resolution - Stress analyses concluded that the CACs were operable in the past regarding structural concerns identified in CR 02-05563. The structural report concluded that, "...Based on the lack of significance or the continued structural acceptability identified with the numerous finding associated with the CAC coil modules and their support structure, the CAC operability assessment is considered to be unaffected by the composite findings of all currently identified, structural-related CAC concerns". The team determined that the licensee appropriately used ASME Code F stress criteria in the structural analysis. This item is closed.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-03e.

RAM Item No. - URI-37 **Closed: Y**

Description of Issue - Issue on whether stem-to-disc separation of SW valve SW-82 was credible and whether stem-to-disc separation was required to be assumed as part of a passive failure analysis.

Description of Resolution - The team determined that valve SW82 was a butterfly valve. Even if stem-to-disc separation occurred, it was extremely unlikely that flow would be blocked. Therefore, the team determined that this failure mode was not credible and did not need to be considered as part of a passive failure analysis. This item is closed.

Reference Material - NRC Inspection Report 05000346/2003010 (ADAMS Accession No. ml040680070) and URI 05000346/2002014-05.

March 22, 2004

RAM Item No. - URI-38 **Closed: Y**

Description of Issue - In November of 2002, the NRC Identified a Potential Concern for Inadequate Over-pressure Protection (OP) for the Containment Air Coolers (CACs), Decay Heat Removal (DHR) Coolers, Emergency Diesel Generator Jacket Water (EDGJW) Heat Exchangers and Associated System Piping (URI No. 06 in Inspection Report 05000346/2002014).

Description of Resolution - On January 23, 2004, the inspectors completed the on-site inspection of URI 05000346/2002014-06. This review was focused on the location of the system relief valves to ensure OP was provided for the CACs, EDGJW heat exchangers and DHR coolers under operating/design basis conditions. This review was prompted by previous NRC questions/concerns for implementation of the Code OP requirements primarily focused on the CACs. For example, the NRC had questioned the use of locked open valves between the relief valve and the Code components requiring relief protection with respect to meeting the Code requirements for positive controls and interlocks on stop valves. These specific requirements and system configurations associated with OP protection were discussed with NRC staff in the Office of NRR and no concerns for Code compliance were identified. Specifically, the inspectors confirmed that:

- The EDGJW coolers and CACs were not Code stamped vessels and thus did not have component level design requirements governing OP protection. The OP protection for the CACs was provided by pressure relief devices for the service water system in which the CACs were installed.

- The DHR coolers were Code stamped vessels, which had component level OP protection requirements from the original design Code (ASME Code, Section III and VIII, 1968 Edition). The inspectors confirmed that the configuration and location of the system OP protection devices was consistent with these requirements.

- For the component cooling water, service water and decay heat removal piping systems which contained these components, the applicable design Code was the ASME Code, Section III, 1971 Edition. This design Code contained specific requirements associated with the location, capacity and types of relief protection required. The inspectors confirmed that the configuration and location of the system OP protection devices was consistent with these requirements for the piping sections containing these components.

For these systems and components, the licensee had not produced a written document that explicitly identified how the applicable OP protection requirements from the design Codes were implemented. Without a written record describing how the Code OP protection requirements were implemented, the inspectors were concerned that changes to the plant design or operation could place these systems/components outside the Code design basis. For example a change in plant operating lineups or system components could render the Code OP protection strategy ineffective and ultimately result in damaged equipment. Based upon this observation, the licensee implemented actions (CR 04-0052) to document the OP protection strategy for these systems and components in controlled safety-related calculations.

In conclusion, the inspectors did not identify any system normal or emergency operating configurations or lineups that would result in isolating the CACs, EDGJW coolers and DHR coolers from OP protection devices, without considering these components and associated piping systems inoperable. Further, no deviations from applicable Code requirements were identified with respect to location of relief protection devices for these components. Therefore, URI 50-346/2002-014-06 is considered closed.

Reference Material - Inspection Report No. 50-346/04-02.

RAM Item No. - URI-39 **Closed: Y**

Description of Issue - Failure to adequately evaluate radiological hazards (White Finding).

Description of Resolution - A supplemental team inspection was conducted in accordance with Inspection Procedure 95002, "Inspection For One Degraded Cornerstone or Any Three White Inputs In a Strategic Performance Area," to assess the licensee's root cause evaluations and corrective actions for the two White findings in the occupational radiation safety cornerstone. In addition, relevant sections of Inspection Procedure 95003, "Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input" were used as guidance during this inspection. The purpose of the supplemental inspection was to: (1) provide assurance that the root and contributing causes for the individual White findings in the occupational radiation safety area and the collective performance which resulted in the degraded cornerstone were understood; (2) independently assess the extent of condition and generic implications of these performance issues; and (3) provide assurance that the corrective actions were sufficient to prevent recurrence. The team concluded that the licensee's root cause evaluations for the White performance issues were completed using systematic techniques, were conducted at the appropriate depth, and adequately identified the primary and contributory causes of the issues. The NRC also concluded that the licensee's corrective action plans were adequate to address the root and contributing causes that were identified in the licensee's evaluation so as to prevent recurrence. Additionally, the team determined that significant progress had been made to improve the licensee's radiation protection program. The licensee's analyses of the White performance issues determined that inadequate work direction and management systems, including problems with radiation protection management oversight, were the root causes of the performance problems, and recent changes have been made in radiation protection management. The team did not identify any significant concerns associated with the current radiation protection program's effectiveness, or significant problems related to the licensee's root cause evaluations for the radiation protection performance problems.

Reference Material - DRS Inspection Report No. 50-346/03-08 (ADAMS Accession No. ml031500693).

March 22, 2004

RAM Item No. - URI-40 **Closed: Y**

Description of Issue - Failure to obtain timely and suitable measurements (White Finding).

Description of Resolution - A supplemental team inspection was conducted in accordance with Inspection Procedure 95002, "Inspection For One Degraded Cornerstone or Any Three White Inputs In a Strategic Performance Area," to assess the licensee's root cause evaluations and corrective actions for the two White findings in the occupational radiation safety cornerstone. In addition, relevant sections of Inspection Procedure 95003, "Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input" were used as guidance during this inspection. The purpose of the supplemental inspection was to: (1) provide assurance that the root and contributing causes for the individual White findings in the occupational radiation safety area and the collective performance which resulted in the degraded cornerstone were understood; (2) independently assess the extent of condition and generic implications of these performance issues; and (3) provide assurance that the corrective actions were sufficient to prevent recurrence. The team concluded that the licensee's root cause evaluations for the White performance issues were completed using systematic techniques, were conducted at the appropriate depth, and adequately identified the primary and contributory causes of the issues. The NRC also concluded that the licensee's corrective action plans were adequate to address the root and contributing causes that were identified in the licensee's evaluation so as to prevent recurrence. Additionally, the team determined that significant progress had been made to improve the licensee's radiation protection program. The licensee's analyses of the White performance issues determined that inadequate work direction and management systems, including problems with radiation protection management oversight, were the root causes of the performance problems, and recent changes have been made in radiation protection management. The team did not identify any significant concerns associated with the current radiation protection program's effectiveness, or significant problems related to the licensee's root cause evaluations for the radiation protection performance problems.

Reference Material - DRS Inspection Report No. 50-346/03-08 (ADAMS Accession No. ml031500693).

March 22, 2004

RAM Item No. - URI-41 **Closed: Y**

Description of Issue - Inappropriate Licensee Notification of NRC Inspector Activity and failure of licensee personnel to enforce an obvious OSHA safety deficiency.

Description of Resolution - The inspector informed licensee management that employees who are aware of safety requirements should enforce those requirements when deficiencies are observed. Additionally, licensee employees should not warn other licensee employees of the NRC inspector's presence as it could leave impression that behavior of licensee individuals was dependent on whether or not the NRC inspector was watching a given activity. Although 10 CFR 50.70(b)(4) requires, in part, that "the arrival and presence of the NRC inspector is not announced or otherwise communicated by its employees or contractors to other persons at the facility unless specifically requested by the NRC inspector," the inspectors determined that the advance notice in this case was not a violation of regulatory requirements. "[10 CFR] Part 50 Statement of Considerations," October 25, 1988, states that "The intent of this rule is to prevent site and contractor personnel from widespread dissemination . . . of the presence of an NRC inspector. It further states that " . . . the NRC expects to reserve enforcement action for significant intentional violations of the rule." The inspectors determined that there was no widespread dissemination of the presence of the NRC inspectors.

A licensee mechanical maintenance person observed the NRC inspectors signing in at the auxiliary building radiation protection access point prior to entering containment. The same person warned the two other licensee employees of the NRC inspectors in containment. In addition, the inspectors determined that there was no significant intentional violation of the rule. The licensee reviewed General Employee Training and found no specific reference to 10 CFR 50.70(b)(4). Because the mechanical maintenance person was not trained on the regulation the inspectors determined that there was not a significant intentional violation of the rule. The licensee took action to include the regulation in General Employee Training.

This issue was discussed in inspection report 2002-017 and documented in the licensee corrective action program as CR 02-9278. In response to this issue, the licensee conducted an independent investigation of the event and conducted site wide training on the requirements of 10CFR50.70(b)(4). The training was completed in November 2002 and the inspectors were briefed on the results of the licensee investigation.

Reference Material - See Inspection Report No. 2002-017, which is in ADAMS at accession no. ml023430380.

March 22, 2004

RAM Item No. - URI-42 **[See RAM Item No. L-90]** **Closed: Y**

Description of Issue - Inadequate Implementation of the Corrective Action Process Which Led to Not Identifying a Potentially Reportable Issue Regarding Containment Air Coolers

Description of Resolution - This issue was reviewed and LER 2002-008 was issued on December 31, 2002. NOP-LP-2001 was revised clarifying the requirement to perform a reportability review. A corrective action was initiated to review all significant Condition Reports issued from January 1, 2002, to November 13, 2002, to ensure adequacy of reportability reviews.

This item is closed. Also see related closure documentation for L-90 below. L-90 was previously closed as documented in Panel meetings on 10/9/03. However, the CATI also reviewed L-90 and provided closure documentation at that time since it was related to this URI.

Reference Material - See Reference Material cited for Closure of RAM Item L-90.

March 22, 2004

RAM Item No. -URI-43 **Closed: Y**

Description of Issue - Final Evaluation of LER 50-346/2002-006-00. See also CR 02-5590 [EDG Exhaust]. Specifically, this item involves resolution of six feet of EDG exhaust stacks which were unprotected against tornado missiles and that portions of a concrete barrier were degraded.

Description of Resolution - The team determined that the licensee had evaluated the non-conforming conditions using a computer code (TORMIS) discussed in Electric Power Research Institute (EPRI) Topical Report NP-2005, "Tornado Missile Risk Evaluation Methodology," Volumes I and II, August 1981. Based on use of this code, the licensee determined the probability of the unprotected areas being struck by a tornado missile was low.

The licensee revised the USAR to incorporate the TORMIS methodology, including a provision which allowed it to be used to evaluate tornado missel and wind loading conditions. Utilizing the TORMIS methodology, the licensee determined that no modifications were necessary to the diesel generator stacks and determined that the diesel generators were operable. Further, the licensee identified that repairs were necessary to other degraded components. The licensee physically repaired the degraded concrete to restore its tornado protection capability.

As part of the USAR change, the licensee performed an evaluation as required by 10 CFR 50.59. During inspector review of this evaluation, the team questioned whether the licensee had appropriately followed the guidance in Nuclear Energy Institute standard NEI 96-07, which NRC endorsed in Regulatory Guide 1.187. The licensee acknowledged that sufficient detail was not provided in its 10 CFR 50.59 evaluation to support the conclusions in the evaluation. The licensee initiated condition report 03-06561 to address the deficiency in the 50.59 evaluation.

The team analyzed this issue and determined that it was of very low safety significance, and concluded that this was not a restart constraint. The team identified a NCV of 10 CFR 50.59, “Changes, Tests and Experiments,” having very low safety significance.

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

March 22, 2004

RAM Item No. - URI-44 **Closed: Y**

Description of Issue - Potential Inability for HPI Pumps to Perform Safety Related Function (see LER 03-02)

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.

RAM Item No. - URI-45 **Closed: Y**

Description of Issue - Failure to Effectively Implement Corrective Actions for Design Control Issues Related to Deficient Containment Coatings, Uncontrolled Fibrous Material and Other Debris (see LER-02-05).

Description of Resolution - This item is based on the followup results of inspection into the licensee's event report 02-05, which is documented as a Yellow finding in Inspection Report 50-346/03-15. The writeup closing this item is identical to that closing the associated LER RAM item, LER-05.

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).