April 24, 2000

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NRC INSPECTION REPORT 50-315/2000007(DRS); 50-316/2000007(DRS)

Dear Mr. Powers:

On March 15, 2000, the NRC completed a special inspection of activities at the D. C. Cook Units 1 and 2 facilities. This inspection addressed your actions to resolve Case Specific Checklist (CSC) Item No. 3A, "Inadequate Design Control Pertaining to Uncontrolled and/or Unintended Changes in Plant Design," CSC Item No. 11, "Resolution of Hydrogen Mitigation System Operability and Materiel Condition Issues," and CSC Item No. 14E, "Electrical Protection Coordination including Fuse/Breaker Control Program Ready for Restart," that were established through the NRC's Manual Chapter 0350, "Staff Guidelines for Restart Approval." The enclosed report documents the results of the inspection.

Based on review of calculations and design changes completed under the revised calculation and modification processes, the team concluded that acceptable quality design control products had been produced. Two examples of inadequate design reviews were identified which indicated a weakness in the implementation of the owner acceptance review process. Your staff implemented prompt action to strengthen this process. Overall, these examples did not indicate a fundamental problem with the design control process at D.C. Cook and the team concluded that adequate design controls existed. Therefore, the NRC considers CSC Item No. 3A closed.

The team considered the Electrical Coordination program assessments and corrective actions comprehensive and determined that they encompassed appropriate facets of the plants electrical design. Based on a review of the completed major corrective actions for Unit 2 and the planned corrective actions for Unit 1, the team concluded that an adequate basis for electrical plant coordination and protection would be established. Therefore, the NRC considers CSC Item No. 14E closed.

The team could not reach a conclusion on the operability of the Hydrogen Skimmer and Air Recirculation (CEQ) system. Although calculations had been completed to determine the flow rates within this system, the actual flow rates for areas serviced by this system such as the ice condenser, steam generator enclosures and pressurizer enclosure were less than design values. The team was told that the design flow values were more conservative than needed to mitigate hydrogen, which is the primary function of this system. However, the calculation which

supports this conclusion had not been completed. Therefore, until your staff completes documentation which demonstrates that the CEQ system can maintain hydrogen with the limits established in 10 CFR 50.44, CSC Item No. 11 will remain open.

Based on the results of this inspection, the NRC has determined that two violations of NRC requirements occurred. The violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII B.1a of the NRC Enforcement Policy. The NCVs are described in the subject inspection report. If you contest the violation or severity level of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/RA/

John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Inspection Report 50-315/2000007(DRS);
50-316/2000007(DRS)

cc w/encl: A. C. Bakken III, Site Vice President
J. Pollock, Plant Manager
M. Rencheck, Vice President, Nuclear Engineering
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/2000007(DRS); 50-316/2000007(DRS)
Licensee:	American Electric Power Company
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	February 7 - March 15, 2000
Inspectors:	Melvin S. Holmberg, Reactor Engineer, Team Leader Andrew Dunlop, Reactor Engineer John Chiloyan, Consultant Raymond J. Cooney, Consultant Richard F. Ely, Consultant Charles R. Jones, Consultant
Approved by:	Gary L. Shear, Chief, Plant Support Branch Division of Reactor Safety

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EXECUTIVE SUMMARY

D. C. Cook, Units 1 and 2
NRC Inspection Report 50-315/2000007(DRS); 50-316/2000007(DRS)

By letter dated September 17, 1999, the NRC transmitted the updated Case Specific Checklist (CSC) for the Donald C. Cook Nuclear Power Plant which identified issues requiring resolution prior to restart of the Cook Plant. This nonroutine inspection focused on the licensee corrective actions for resolution of CSC Item No. 3A, "Inadequate Design Control Pertaining to Uncontrolled and/or Unintended Changes in Plant Design" (and associated Enclosure 2 Items C.3.2.e & C.3.2.f), CSC Item No. 11, "Resolution of Hydrogen Mitigation System Operability and Materiel Condition Issues," CSC Item No. 14E, "Electrical Protection Coordination including Fuse/Breaker Control Program Ready for Restart," identified in Enclosure 1 of this letter. The standards applied to evaluate the acceptability for resolution of these CSC Items were those described in paragraphs C.1.2, "Corrective Action Development," and C.1.3, "Corrective Action Plan Implementation and Effectiveness," of Enclosure 2 of the NRC letter transmitting the CSC.

Open items identified in NRC inspection reports and Licensee Event Reports requiring inspection/resolution prior to restart of the Cook Plant have been identified in the Restart Action Matrix (RAM) approved by the NRC Manual Chapter 0350 Oversight Panel. In the RAM, open items were identified with designated inspection priorities. The higher priority inspection issues received a more in-depth review during this inspection. Based on adequate corrective actions for resolution of items selected for the more in-depth review, reasonable assurance exists that corrective actions for the similar lower priority inspection issues are adequate. The intent of selecting a sample of items for more in-depth review was to improve NRC efficiency in assessing the restart readiness of the plant and to ensure an appropriate focus on the issues most important from a safety and risk perspective.

<u>Case Specific Checklist Items</u>

- Based on review of calculations and design changes completed under the revised calculation and modification processes, the team concluded that acceptable quality design control products had been produced. Two examples of inadequate design reviews were identified which indicated a weakness in the implementation of the owner acceptance review process. The licensee implemented prompt action to strengthen this process. The team also identified examples of problems which illustrated administrative weaknesses in implementation of the calculation and design control process. Overall, these examples did not indicate a fundamental problem with the design control process at D.C. Cook and the team concluded that adequate design controls existed. Therefore, the NRC considers CSC Item No. 3A and Enclosure 2 Item C.3.2.e and Item C.3.2.f closed (Section E8.1).

- CSC Item No. 11 is open. The team could not reach a conclusion on the operability for the Hydrogen Skimmer and Air Recirculation (CEQ) system. The calculations completed for this system demonstrated that flow rates for areas serviced by this system such as the ice condenser, steam generator enclosures and pressurizer enclosure were less than design values. The accuracy of system design flows could not be verified,

because the calculations did not describe the impact of test data and model uncertainty. The team was told that the design flow values were more conservative than needed to mitigate hydrogen which is the primary function of this system. However, the calculation which supports this conclusion had not been completed. Therefore, until the licensee completes documentation which demonstrates that the CEQ system can maintain hydrogen within the limits established in 10 CFR 50.44, CSC Item No. 11 will remain open (Section E8.2).

- The team considered the Electrical Coordination program assessments and corrective actions comprehensive and determined that they encompassed appropriate facets of the plants electrical design. Based on a review of the completed major corrective actions for Unit 2 and the planned corrective actions for Unit 1, the team concluded that an adequate basis for electrical plant coordination and protection would be established. Therefore, CSC Item No. 14E is closed (Section E8.3).

Restart Issues

- Nine high priority restart issues from the Restart Action Matrix were examined in detail to ensure that the licensee properly addressed the issues and completed corrective action, or had them scheduled and tracked for completion prior to plant restart. Seven of the nine, discussed in this report, were considered satisfactorily completed. The remaining two required additional actions to resolve the issues and consequently could not be closed (Sections E8.4 - E8.8).

- Nine low priority issues from the Restart Action Matrix were examined to verify that: the issues were entered in the corrective action system; the issues were properly characterized and classified; appropriate corrective actions had been specified; and the corrective actions were scheduled and tracked. Seven of these issues listed in the report are closed (Section E8.9).

Engineering Issues

- A non-cited violation was issued for the licensee's failure to perform an adequate design review and verify the analysis inputs for the steam generator margin to overfill analysis (Section E8.1).

- The team identified a weakness associated with the lack of design basis documentation available to address the 4KV motor cable protection against short-circuits (Section E8.3).

- A non-cited violation was issued for the licensee's failure to perform an adequate design review and identify the correct relay function in an electrical containment penetration protection calculation (Section E8.3).

- The action plan for resolution of the emergency diesel generator air systems did not specify the actions to be taken to resolve the operability for Unit 1 emergency diesel generator air systems (Section E8.4).

- The revised sump action plan, revised specification and procedure controlling fibrous materials in containment were adequate to resolve previously identified concerns in these areas (Section E8.5).

- The team determined that the overall level of effort being applied to upgrade the licensee's High Energy Line Break program was substantial and consistent with that needed to resolve the deficiencies. The licensee's upgrade program included reanalysis, revised program documentation, and physical plant changes. Overall, the corrective actions implemented by the licensee adequately addressed the High Energy Line Break related deficiencies (Section E8.6).

- The team questioned the licensee's basis for not performing a post modification hot plant check of the pressurizer surge line pipe whip restraint gaps and considered this issue an Inspection Followup Item (Section E8.6).

- The team identified insufficient documentation for the flowrate used in confirming the absence of vortexing at the lowest useable refueling water storage tank volume (Section E8.7).

- The team determined that corrective actions to install dedicated containment level instrumentation were adequate to resolve concerns for sump recirculation level uncertainty (Section E8.8).

Report Details

Background

Both units have been in an extended shutdown since September 9, 1997.

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Design Control (CSC Item No. 3A) and Corporate Engineering Support (Items C.3.2.e and C.3.2.f)

a. Inspection Scope

The NRC identified as CSC Item No. 3A, a concern for inadequate design control pertaining to uncontrolled and/or unintended changes in the plant design. In Enclosure 2 of the NRC letter transmitting the CSC, the NRC identified a need to evaluate the effectiveness of corporate engineering support and the effectiveness of the design modification process (Items C.3.2.e and C.3.2.f). To assess these performance areas, the team reviewed approximately 28 calculations and 17 design changes completed under the recently revised calculation and design change processes. Additionally, the team reviewed the engineering technical basis for deferral of plant modifications until post restart.

b. Observations and Findings

b.1 Design Control Root Cause Assessment

In condition report (CR) 99-00594, the licensee documented the root causes for the breakdown of design control and configuration management. The licensee performed an analysis of 147 findings from ten self-assessments of the design organization's programs and 24 findings from the Design Engineering Management Assessment Report dated December 12, 1998. The overall root cause was that "previous senior management teams failed to recognize that maintaining the design basis and providing a strong configuration management are vital functions in nuclear power operations. This resulted in strategic errors, low expectations, and low commitment to controlling the CNP (Cook Nuclear Plant) design basis." The specific causes were divided and grouped as follows: seven "management causes;" seven "organizational contributing causes;" and four "programmatic contributing causes." The evaluation performed considered the root cause of design control deficiencies in an aggregate manner, vice determining the exact cause of individual design errors. Thus, no documentation existed to provide a direct correlation between NRC and licensee identified problems with design control to the 18 specific root/contributing causes. The licensee "judged" that six of these root causes were directly applicable to the uncontrolled/unintended design change area and incorporated these into Restart Action Package (RAP) 3, "Uncontrolled/Unintended Plant Design Changes." The team concluded that these root

causes appeared to bound the identified deficiencies related to uncontrolled/ unintended design changes.

b.2 Design Control Corrective Actions

Development and implementation of the licensee's corrective action plan for RAP 3A had been previously evaluated by the NRC and documented in inspection report 50-315/99029(DRS); 50-316/99029(DRS) and were considered acceptable. However, insufficient engineering products had been generated under the revised design control program to evaluate the effectiveness of the new process. Therefore, the team assessed the effectiveness of the new design process by evaluating the calculations and modifications reviewed herein as discussed in the following section.

b.3 Effectiveness of the Calculation and Modification Process

The team identified a weakness in the implementation of the owners acceptance review process. Two examples were identified in which the licensee failed to complete adequate design reviews during the owners acceptance review process. One example involved an electrical protection calculation used to demonstrate electrical penetration integrity (Section E8.3.b.3) and the second example involved the margin to overfill analysis used to validate original licensing basis assumptions in the steam generator tube rupture (SGTR) event (Section E8.1.b.5). The team noted that the licensee took prompt action to strengthen the owners acceptance review process (Section E8.1.b.5).

The team also identified examples of problems which illustrated administrative weaknesses in implementation of the calculation and design control process. The team identified an unrestricted vacuum calculation that relied on information from a restricted structural calculation (Section E8.7) and the team identified that the documented basis for flow values selected to resolve level uncertainties associated with minimum refueling water storage tank (RWST) tank level was weak (Section E8.7). The team also identified that design documentation for the 4 kilo-volt (KV) motor conductor cable protection (Section E8.3.b.4.1) had not been established.

Overall, these examples did not indicate a fundamental problem with the design control process at D. C. Cook and the team concluded that adequate design controls existed. Therefore, based on review of calculations and modifications discussed in the following sections, the team concluded that acceptable quality design control products had been produced, and CSC Item No. 3A and Enclosure 2 Items C.3.2.e and C.3.2.f are closed.

b.4 Adequate Technical Basis for Deferral of Modifications to Post Restart

The team reviewed the technical basis documented in the System Index Data Base System (SIDS) for eight Unit 2 (or common) system design changes which had been canceled or deferred until after Unit 2 startup. The licensee's process for control of deferred or canceled design changes was described in Attachment 11, "Scope Change Form," of PMP-7200.RST.004, "Expanded System Readiness Review Program." The information in this form was not consistently completed for the deferred or canceled design changes reviewed by the team. In seven cases, the block on the form for technical justification for the scope designation or change was marked "see attached"

without specific reference to an attachment number. The licensee stated that "the attached" in each case referred to the justification documented in the comments portion of the SIDS data base. The team considered this documentation less than rigorous, however, no technical concerns were identified by the team for the justifications provided.

b.5 Steam Generator Margin To Overfill Analysis

The team reviewed AEP 99-486, "American Electric Power Donald C. Cook Nuclear Plant Unit 2 Supplemental Steam Generator Tube Rupture Analysis." This analysis was performed to confirm that operator actions could be accomplished in time to prevent overfill of the steam generator (SG) during a tube rupture event. If the SG was overfilled during this event, damage could occur to the main steam piping or relief valve potentially creating off-site release of radioactive material greater than what had been previously assumed. A draft version (SAE-CRA-99-316) of AEP 99-486 was used to establish the basis for the acceptable operator response time requirements for the SGTR scenario in the emergency operating procedure (EOP) E-3, "Steam Generator Tube Rupture." The AEP 99-486 analysis was also used to confirm original licensing basis assumptions that the SG does not overfill during the SGTR event.

To confirm the margin to SG overfill for Unit 2, the licensee had contracted with Westinghouse to perform a detailed thermal-hydraulic analysis. The results of this analysis were documented in analysis report AEP 99-486, dated December 20, 1999. This report was not subject to a review by the licensee's Design Review Board, however, the licensee performed an owners acceptance review (OAR) on December 28, 1999. The team noted that the licensee failed to identify the lack of a specific reference to the actual Westinghouse calculation which formed the basis for AEP 99-486. During this review, the licensee had also failed to identified the lack of documentation or reference to many of the plant parameters used as inputs in the analysis which formed the basis for the conclusions of this report. Examples of inputs not explicitly documented in the report included the SG tube break flow area, condensate storage tank temperature, initial reactor coolant system temperature and pressure. Westinghouse had reportedly selected nominal values or bounding inputs to perform the calculation used as the basis for this report. However, no basis was documented for when nominal or bounding calculation inputs were selected. The team's questions into the input values used prompted the licensee to identify and document errors in steam generator pressures used to calculate the auxiliary feed water (AFW) system flowrates and the use of a nonconservatively high value for T_{ave} as analysis inputs. The licensee documented these errors in CR 00-02975. The failure to control and verify appropriate analysis input assumptions for the calculation which formed the basis for AEP 99-486 represented an inadequate design review as discussed below.

10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews. In this case, the OAR of AEP 99-486 completed on December 28, 1999, was inadequate to verify the adequacy of the plant design because design inputs to this analysis had not been verified in this review. Failure to perform an adequate design review for AEP 99-486 and verify the analysis inputs is a violation of 10 CFR 50, Appendix B, Criterion III. However, this Severity Level IV violation is being

treated as a Non-Cited Violation (50-316/2000007-01(DRS)) consistent with Section VII B.1a of the NRC Enforcement Policy. The inadequate owners acceptance review for this issue was documented in CR 00-03212.

The OAR process was described in procedure 12-EHP-5043.OAR.001, "Owners Acceptance Review," and consisted of completing a 15 question checklist. The two contracted licensee personnel who performed the OAR review of AEP 99-486 were questioned by the team. They had each received training on the OAR process and believed that the review for AEP 99-486 was completed as the procedure intended and that a Design Review Board review was not required. The licensee initiated CR-00-3208 to document the unclear expectation for the Design Review Board review of Westinghouse Letter Reports. A similar condition had been previously documented in CR 99-23625, in which the licensee had identified that the OAR process was not consistently understood and applied. As an immediate corrective action, the Design Engineering Director issued a memorandum dated February 28, 2000, which required engineering personnel to obtain supporting calculations and analysis to complete OARs for all letter reports and similar documents. Further, this memorandum required the report and supporting documents to go through a Design Review Board unless a waiver is obtained. The team concluded that this issue had been adequately captured in the corrective action system and that the licensee had initiated appropriate corrective actions to strengthen this process weakness.

b.6 Auxiliary Feedwater System Analysis

The team reviewed calculation MD-12-AFW-001-N, "AFW [Auxiliary Feedwater] System Analysis for Loss of AC and Main Steam Line Break," which was performed as a corrective action from the licensee's calculation reconstitution initiative. This calculation was performed to confirm the minimum and maximum AFW system flowrates used as inputs to analysis WCAP-14286 for a loss of offsite power and main steam line break scenarios. The team identified that the licensee had used the differential pressure for the original 1/32 inch mesh size AFW system strainers, vice using the differential pressure associated with the 1/8 inch mesh size strainers installed under design change package (DCP) 2 DCP-667, "AFW System Suction from ESW [Essential Service Water]." The licensee documented this error in CR 00-02725 and concluded that this error would have little impact on the analysis. The team verified the inputs to this calculation such as the flow retention system setpoints which conservatively bound the actual measured system flows recorded in the latest completed system surveillance test 12 EHP 6040 PER.116, "AFW Flow Retention," completed June 12, 1997. Overall, the calculation assumptions, design inputs and acceptance criteria were clearly documented, conservative, and technically supportable.

c. Conclusions on Design Control (CSC Item No. 3A) and Corporate Engineering Support (Items C.3.2.e and C.3.2.f)

Based on review of calculations and design changes completed under the revised calculation and modification processes, the team concluded that acceptable quality design control products had been produced. Two examples of inadequate design reviews were identified which indicated a weakness in the implementation of the OAR process. The licensee implemented prompt action to strengthen this process. The

team also identified, examples of problems which illustrated administrative weaknesses in implementation of the calculation and design control process. Overall, these examples did not indicate a fundamental problem with the design control process at D.C. Cook and the team concluded that adequate design controls existed. Therefore, the NRC considers CSC Item No. 3A and Enclosure 2 Item C.3.2.e and Item C.3.2.f closed.

A non-cited violation was issued for the licensee's failure to perform an adequate design review and verify the analysis inputs for the steam generator margin to overfill analysis.

E8.2 (Open) Resolution of Hydrogen Mitigation System Operability (CSC Item No. 11)

a. Inspection Scope

The NRC identified as CSC Item No. 11, concerns associated with hydrogen mitigation system operability and material condition. To evaluate licensee performance in resolving operability of the hydrogen skimmer and air recirculation (CEQ) system, the team reviewed supporting calculations, test data, and other completed corrective actions documented in RAP 11 "Resolution of Hydrogen Mitigation System Operability and Material Condition Issues."

b. Observations and Findings

b.1 Hydrogen Mitigation System Corrective Actions

The NRC had previously reviewed RAP 11, Revision 0 in Inspection Report 50-315/99029(DRS); 50-316/99029(DRS) and identified a deficiency in that the RAP did not include CR 99-25414 and its associated action items. The licensee explained that the focus of restart action plans was to address root causes of the condition and because CR 99-25414 and reference CRs 99-12872 and 99-12875 did not require root cause analysis, they had not been included in Revision 0 of RAP 11. The licensee stated that this was an error in judgment and corrective actions of CR 99-25414 were subsequently incorporated into RAP 11. The team considered that Revision 2 of RAP 11 documented an acceptable approach to resolution of this issue. However, this issue could not be closed due to lack of completed documentation to demonstrate the system's capability to mitigate post accident hydrogen levels as discussed in Section E8.2.b.3.

b.2 Hydrogen Mitigation System Modifications

b.2.1 CEQ Fan Logic and Time Delay 2-DCP-0650

A completed corrective action for the CEQ system included design change 2-DCP-650, Revision 0, "CEQ Fan Logic and Time Delay," which had been previously reviewed by the NRC and found to adequately translate the new fan start signal timing assumed in the analysis into the plant design. In this design change, the licensee revised the start of fan logic from hi-hi containment pressure setpoint to the hi containment pressure setpoint which reduced the start time of the CEQ fans from nine minutes to two minutes to facilitate earlier ice melt and ensure sufficient containment sump level for small break loss-of-coolant-accidents (LOCAs). The DCP affected several systems (e.g., cooling water supplies, the solid state protection system) and the licensee had identified the

required UFSAR revisions, license amendments, and design basis document changes. This DCP was comprehensive and detailed in the description of the design change impact on other plant systems.

b.2.2 HV-CEQ-1(2) Fan Power Cable Replacement 2-DCP-4476

The team reviewed design change 2-DCP-4476 HV-CEQ-1(2), “Fan Power Cable Replacement.” This DCP provided larger power cables to the CEQ fan motors to assure sufficient voltage at the motors with under voltage condition on the 4KV bus. The cable size selected was sufficiently large, such that it could accommodate any fluctuations in voltage associated with changes to the existing 4KV degraded voltage setpoint. The team considered this an appropriate resolution to address potential degraded voltage concerns for the power to CEQ fan motors.

b.3 Hydrogen Mitigation System Calculations

The team reviewed calculations MD-2-HV-011-N, “Donald C Cook Unit 2 Containment Air Recirculation/ Hydrogen Skimmer System Pressure Drop Calculation,” and MD-2-HV-012-N, “Unit 2 Containment Air Recirculation/ Hydrogen Skimmer System CEQ Fan Pressure Drop Comparison and Motor Evaluation for fans ½-HV-CEQ-1/2." The team identified weaknesses in these calculations. The scope of MD-2-HV-011-N stated that it applied to Unit 2 and Unit 2 test data was applied to a Unit 1 piping model. In MD-2-HV-012-N the licensee stated that MD-2-HV-011-N was applicable to Unit 1 and that it’s objective was to demonstrate the results were applicable to Unit 2. Neither calculation established the technical acceptability of using a Unit 1 piping model and Unit 2 test data. Further, in the general conclusions of MD-2-HV-011-N a note stated that no adjustment was made for uncertainty with respect to the test data. Therefore, the team was not able to assess the accuracy of system flowrates derived in these calculations, because the calculations did not describe the impact of test data and model uncertainty. The licensee documented the team’s observations of inconsistencies for calculations MD-2-HV-011-N and MD-2-HV-012-N in CR 00-02673 and CR 00-02396. These condition reports were subsequently added to RAP 11.

Based on calculation MD-2-HV-011-N, flow rates for areas serviced by the CEQ system such as the ice condenser, steam generator enclosures, and pressurizer enclosure were calculated to be less than design flow values. The team was told that the design flow values were more conservative than needed to mitigate hydrogen, which is the primary function of this system. However, the calculation which supports this conclusion (FAI 99-77) had not been completed. To evaluate the licensee’s planned technical approach for FAI 99-77, the team reviewed a calculation FAI 99-55, “Hydrogen Distribution in the D. C. Cook Containment Under Degraded Containment Air Recirculation and Hydrogen Skimmer-Flows for Justification of Past Operations,” used to justify past system operability. The team identified that the licensee had chosen a potentially non-conservative interpretation of 10 CFR 50.44 requirements with respect to assumptions for hydrogen generation.

Section (d)(1) of 10 CFR 50.44 required, in part, that the contribution of hydrogen contributed by the core metal-water reaction be five times the total amount of hydrogen calculated in demonstrating compliance with 10 CFR 50.46. The licensee had

documented in FAI 99-55 that if this assumption was used with the new analytical methods, transient hydrogen concentrations of greater than four volume percent (with oxygen concentration above five volume percent) were calculated to occur in reactor coolant loop and ice condenser subcompartments during the early stages of a LOCA. The licensee had rationalized that the Section (d)(1) requirement applied to the global containment post LOCA environment and not to the subcompartment hydrogen levels. It appeared that the licensee may need to seek formal NRC review and approval to use this interpretation of 10 CFR 50.44 requirements. The licensee was reportedly considering this option for the final hydrogen calculation FAI 99-77 to be used in resolution of this issue.

c. Conclusions on Resolution of Hydrogen Mitigation System Operability

The team could not reach a conclusion regarding the operability of the CEQ system based on review of licensee documentation. The calculations completed for this system demonstrated that flow rates for areas serviced by this system such as the ice condenser, steam generator enclosures and pressurizer enclosure were less than design values. The accuracy of system design flows could not be verified because the calculations did not describe the impact of test data and model uncertainty. The team was told that the design flow values were more conservative than needed to mitigate hydrogen which is the primary function of this system. However, the calculation which supports this conclusion had not been completed. Therefore, until the licensee completes documentation which demonstrates that the CEQ system can maintain hydrogen with the limits established in 10 CFR 50.44, CSC Item No. 11 will remain open.

E8.3 (Closed) Electrical Protection Coordination Program Ready for Restart (CSC Item No. 14E)

a. Inspection Scope

The NRC identified as CSC Item No. 14E, a concern for the Electrical Protection Coordination including Fuse/Breaker Control program ready for restart. To evaluate licensee performance in this area, the team reviewed design change packages, supporting calculations, and other completed corrective actions as documented in Assessment SA-1999-008-NED, “Electrical Protection Program,” Revision 2, and Assessment SA-1999-002-NED, “Electrical Load Control Program,” Revision 2.

The licensee had previously identified that underrated fuses had been used in the 250 VDC system which could result in lack of protective coordination and reported this condition to the NRC in Licensee Event Report (LER) 50-316/99027-00. This issue was incorporated into the NRC RAM and classified as a high priority inspection Item R.2.14.1. The team reviewed the licensee’s root cause determination and corrective actions documented in CR 99-19746 and DCP-4392 for this issue.

b. Observations and Findings

b.1 Electrical Protection and Load Control Program Assessments (SA-1999-002-NED) and (SA-1999-008-NED)

The licensee performed assessments of the Load Control and Electrical Protection programs as documented in SA-1999-002-NED and SA-1999-008-NED. Based on these assessments, the licensee determined that there was no overall control of loads being added to the systems and that protection of the equipment was not coordinated. To address this concern, the licensee established the Load Control program. The stated purpose of the licensee's Load Control program included determination of potential short circuit currents, voltages available at buses throughout the systems and coordination of protective devices to assure that the breaker or fuse immediately upstream of a fault actuates first to isolate the fault. The purpose of the Load Control program also included ensuring that electrical protection coordination is adequately maintained following plant modifications. The team considered that these assessments were comprehensive and encompassed appropriate facets of the plant's electrical design.

The licensee's proposed corrective actions consisted of reviewing and revising electrical protection and load control calculations. For the 250 VDC systems, the corrective actions included review of the fuse selection program and replacement of approximately 900 fuses in the Unit 2 circuits. A similar review and replacement program was planned for Unit 1. For the 4 kilovolt (KV) AC, 600V alternating current (AC) and 120VAC electrical systems, the corrective actions consisted of developing load flow, short-circuit, voltage drop, breaker and relay setting calculations by incorporating field verified design inputs. A sample of the calculations or design changes completed as corrective actions for this program were reviewed by the team and discussed in the following sections. Based on a review of the completed major corrective actions for Unit 2 and the planned corrective actions for Unit 1, the team concluded that an adequate basis for electrical plant coordination and protection would be established.

b.2 Electrical Protection Modification 2-LDCP-4369

The team reviewed design change 2-LDCP-4369 "400A [ampere] Fuses and Fuse Holder Block on Switchgear 2-21D3," which documented the installation of the three 400A fuses applied as backup protection for the 350MCM conductors in containment penetration No. 2-CEP-3P3. Based on the completed corrective actions and actions committed to be completed prior to startup, the team concluded that this design change had adequately addressed the dual protection requirement for the affected containment electrical penetration assemblies.

b.3 Electrical Containment Penetration Protection Calculation 2-E-N-PROT-PEN-001/0

The team reviewed calculation Electrical Containment Penetration Protection 2-E-N-PROT-PEN-001/0 and supporting documents, including design information transmittal (DIT)-B-00431-00. The purpose of this calculation and DIT was to demonstrate that the containment electrical penetration protection assemblies were adequately protected by two independent protection systems against short-circuits to ensure containment

integrity. Containment integrity was demonstrated by comparing the penetration capability against the operating characteristics of the associated containment penetration protective devices associated with the 4KV, 600V and 480V containment penetration circuits. In DIT-B-00431-00, the licensee addressed the adequacy of electrical protection associated with low voltage control and instrumentation containment penetration circuits.

To provide ground backup protection to the reactor coolant pump (RCP) feeder, the team identified that the type IAC52 ground relay on the 4KV bus would provide fault protection for the containment penetration. The team selected the type IAC52 ground relay, based on its setpoint of 47 cycles at three times pickup, which indicated a time dial of approximately 1.6. Thus, at the maximum available ground-fault current of 2000A, or 3.33 times pickup, it would trip in 0.77 second. The calculation had not established the magnitude of available ground fault current and; therefore, the team calculated the available ground fault current to be 2000A. The team based the available ground fault current on the grounding resistor data given in D. C. Cook Nuclear Plant Unit No. 2 Drawing No. E-1000B-12/10, "One Line Diagram."

The team identified that in calculation 2-E-N-PROT-PEN-001/0, the licensee had incorrectly credited a ground instantaneous relay PJC11 as a tripping relay on the incoming feeder to the 4KV buses to provide backup protection for down stream feeder ground faults. Specifically, Section 7.1.1 of calculation 2-E-N-PROT-PEN-001/0 stated, "... If the RCP feeder breaker fails to operate, ground fault relay PJC11 given on incoming feeder to 4.16KV Bus set to operate at 300 Amperes (as given in Section 2.1.8) will operate and provide the back-up protection by opening the Incoming Breaker." Contrary to this description, the type IAC52 ground relay would actuate (based on calculations performed by the team as discussed above) instead of the PJC11 instantaneous ground relay to perform the fault protection function. This calculation had been prepared by a contractor and the licensee had completed an OAR on February 8, 2000, and failed to identify the error.

10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews." In this case, the OAR of calculation 2-E-N-PROT-PEN-001/0 completed on February 8, 2000, was inadequate to verify the adequacy of the plant design because the licensee review failed to verify the correct relay function. Failure to perform an adequate design review for this calculation and identify the correct relay function is a violation of 10 CFR 50, Appendix B, Criterion III. However, this Severity Level IV violation is being treated as a Non-Cited Violation (50-316/2000007-02(DRS)), consistent with Section VIII B.1a of the NRC Enforcement Policy. The licensee documented this issue in CR 00-03233.

In Calculation No. 2-E-N-PROT-PEN-001/0 and DIT-B-00431, the licensee demonstrated two independent protection systems for the protection of electrical containment penetration assemblies, where the potential exists for a fault inside containment to result in a penetration seal failure, consistent with Regulatory Guide 1.63 and IEEE Standard 741 requirements. Therefore, the team concluded adequate corrective actions had been implemented for this area.

b.4 4KV Safety Related Motor Phase Instantaneous Relay Setting Calculation 2-E-N-PROT-RLY-002/0

The team reviewed the 4KV safety related motor phase instantaneous PJC relay setting calculation 2-E-N-PROT-RLY-002/0 and supporting documents. The purpose of this calculation was to establish the trip settings for Unit-2 safety related 4KV motors phase instantaneous overcurrent relays (General Electric type PJC relays). The team noted that the instantaneous overcurrent relay settings for the 4KV safety related motors, required instantaneous overcurrent relay pickup to be set at 1.75 times the motor locked rotor current to ensure that the relay can provide fault protection without nuisance tripping on high motor starting transient currents. This approach was in accordance with IEEE Standard 242-1986, "IEEE Recommended Practice for the Protection and Coordination of Industrial and Commercial Power Systems." The team concluded that calculation 2-E-N-PROT-RLY-002/0 had used credible field verified and vendor provided equipment specifications, applied adequate acceptance criteria associated with the effects of current transformer saturation on relay pickup settings, and adequately established PJC relay pickup settings.

The team could not review the field implementation measures for the relay setting changes derived in calculation 2-E-N-PROT-RLY-002/0, because the licensee had not yet completed its own review of the relay setting sheets. The team noted that the licensee used relay setting sheets rather than design change package to implement relay setting changes which did not impact equipment modification as allowed by plant procedure 12-EHP-5040-001/1.

b.4.1 Lack of Design Documentation For the 4KV Motor Conductor Cable Protection

The team identified that the licensee lacked design basis documentation to address the 4KV motor conductor cable protection available to mitigate affects of short-circuit currents. The team noted that none of the protection calculations had evaluated the 4KV motor conductor protection against available short-circuit faults. The team used the calculated maximum available fault currents given in calculation 2-E-N-ELCP-4KV-001/0 and estimated that the maximum available fault currents at the motor terminals of the component cooling water (CCW) and safety injection (SI) pump motors. The team estimated that fault currents could exceed the No. 2 American Wire Gage (AWG) aluminum motor feeder cable capability to withstand cable damage from a postulated short-circuit for these motor conductors. The licensee agreed with the team's assessment that the CCW and SI pump motor conductors could be damaged in the event of certain motor circuit faults, but stated that damage would be limited to the single conductor, since the 4KV cables supplying safety related loads have dedicated individual conduits.

On February 25, 2000, the licensee proved their position on this issue as follows:

- The consideration for the capability of the pump feeder cable to withstand short-circuit currents was not a design basis consideration for the Cook Nuclear Plant.

- The consideration of cable damage due to maximum fault current is considered to be an economic concern not a safety concern. Protection for the bus is

provided by the upstream breaker. The design of the buses is such that no single failure/fault can take out the redundant trains of safety-related equipment.

- Even though the cable with #2 AWG aluminum conductors may be damaged by a fault current and would be required to be replaced, the 4KV cables are routed in conduit by themselves from the switchgear to the motor and are not routed in a tray at any point.

- Cook Nuclear Plant also requires that for any cable damage the entire cable must be replaced because repairs are not allowed on a faulted cable. Also, if the end device (e.g., a motor) is faulted, then the cable supplying the device must be tested for damage prior to returning the equipment to service.

On February 29, 2000, the licensee initiated CR 00-03394 to incorporate the D. C. Cook 4KV cable protection philosophy as part of design basis documentation. Additionally, the team was provided a copy of excerpts from an NRC inspection report 50-315/90012; 50-315/90012, in which the NRC reviewed this issue and determined that it posed no immediate safety concern because loss of a single piece of safety equipment would not cause damage to redundant equipment.

b.5 Underrated Fuses Used in 250 VDC System LER 50-316/99027-00 (RAM Item No. R.2.14.1)

During the Expanded System Readiness Review (ESRR) evaluation of the 250 VDC protective devices, the licensee identified that the 250 VDC system contained fuses rated below the 250 VDC nominal system voltage. These fuses may not have provided adequate fault protection at the voltage levels encountered in the system. In a document prepared by a contract engineering firm, “Fuse Applications in DC Systems,” a recommendation was made to the licensee that all fuses which could not show satisfactory performance at 280 VDC be replaced. In October of 1999, the licensee had additional fuse testing performed by one of the fuse manufacturers at 280 VDC utilizing the testing standard UL 198L. The results demonstrated failures in the 30A and 60A fuse casing size families.

The licensee addressed this issue in CR 99-19746. The cause of the condition was attributed to lack of proper evaluation and consideration of available industry operating experience regarding fuse applications and weaknesses in understanding the design and licensing basis of the 250 VDC system. The licensee proposed several corrective actions, the first of which was to perform a fuse coordination study (2-E-N-ELCP-250-001) to assure proper operation during all modes of plant operation. The second action was to replace all safety related fuses rated 60A or less (2-DCP-4392). Other actions included development of a Unit 1 fuse program similar to the Unit 2 program. The team reviewed calculations and design changes produced in response to the corrective actions for this issue as discussed in the following sections. Based on review of the corrective actions which have been implemented, RAM Item No. R.2.14.1 is closed.

b.5.1 250 VDC Fuse Replacement Project 2-DCP-4392

The team reviewed 2DCP-4392, “250 VDC Fuse Replacement Project,” which replaced the safety related fuses as required by corrective actions designated in CR 99-19746. The team observed replacement of several fuses under this DCP and subsequent operation of the circuits after the replacements were made. No anomalies were noted. The team identified no technical or administrative errors in this DCP and considered that it provided an appropriate technical solution for this issue.

b.5.2 250 VDC Battery to Bus Fuse Replacement 2-DCP-0658

The team reviewed CR 99-04815 documenting deficiencies with the output fuses of the battery chargers. The fuse sizes originally installed were rated 300A and the battery chargers have current limiting ratings of 375A. After a station blackout event, the battery chargers could operate at 375A for an extended period of time. To correct this condition and avoid tripping off the battery chargers, the licensee issued 2-DCP-0658 which required replacement of the 300A fuses in the charger transfer switch cabinets with fuses rated at 400A. The fuse vendor performed a coordination test to assure that 600A fuses at the battery disconnect switches would coordinate with the new 400A fuses. The test results indicated that to assure coordination with the new 400A fuses, the 600A fuses in the battery disconnect would have to be the same type as the 400A. The licensee had incorporated this requirement into 2-DCP-0658 which also replaced the existing Bussman Co. type FRS 600A fuses with Ferraz Shawmut type TRS 600A fuses. The team considered that the corrective actions implemented in 2-DCP-0658 provided an appropriate technical solution to this issue.

b.5.3 Calculations Supporting Fuse Ratings

The team reviewed four major DC calculations which support the revised DC fuse ratings:

2-E-N-ELCP-250-006 “250 VDC Battery 2CD System Analysis”
2-E-N-ELCP-250-007 “250 VDC Battery 2AB System Analysis”
2-E-N-ELCP-250-008 “250 VDC battery 2N System Analysis”
2-E-N-ELCP-250-001 “250 VDC System Coordination Study”

Each of the system analysis calculations used a composite duty cycle to size the battery. For the 2CD and 2AB batteries, the composite duty cycle was made up of the first minute of a LOCA /Loss of Offsite Power (LOOP) duty cycle followed by the last three hours and fifty-nine minutes of the station blackout duty cycle. The 2N battery was sized to support operation of the turbine driven auxiliary feed pump for as long as the steam supply to the turbine is available. The licensee conservatively assumed that the steam supply is available for the entire event and used a composite duty cycle for the 2N battery based on the four hour station blackout (SBO) and the three hour LOCA/LOOP. The team confirmed that the assumptions and inputs used in each of these calculations were reasonable.

The system analysis calculations were completed using validated software which performs battery sizing in accordance with IEEE-485-1997, battery charger sizing, short

circuit currents throughout the system being studied, voltages to the end devices for battery 2N and to a representative sample of end devices for batteries 2AB and 2CD. In addition, a database was created for each system being studied to control load data used in the analysis. In calculation 2-E-N-ELCP-250-008, the licensee demonstrated for 2N battery that, voltage criteria is met for all devices and no further analysis is required. All three system analysis calculations demonstrated that short circuit currents available at each node were less than 20,000A and that the protective devices were satisfactory. All three system analysis calculations show that the battery chargers were correctly sized to meet recharging criteria.

The system analysis calculations demonstrated that batteries 2AB and 2N are adequately sized for the composite duty cycles, using a five percent spare capacity as required by the SBO rule for plants of the DC Cook vintage and using a 1.25 aging factor. This aging factor will allow operation of the batteries for their duty cycle until the battery reaches 80 percent capacity. Technical Specification (TS) surveillance requirement 4.8.2.3.2 part e required that at least once per 60 months battery capacity must be shown to be at least 80 percent of the manufacturers rating.

IEEE standard 450 stated that the recommended practice is to replace the battery when it gets to 80 percent capacity when a 1.25 aging factor is used. It also states where a lesser aging factor is used battery replacement is required before 80 percent capacity is reached. The calculation for battery 2CD used a five percent spare capacity and a 1.03 aging factor which will allow operation of the battery to approximately 97.1 percent of the rated capacity in the future. The team reviewed the latest battery capacity test data for battery 2CD recorded in September of 1999, and noted that it was at 103.99 percent and therefore met TS surveillance requirements for battery operability. The licensee chose not to use the 1.25 percent aging factor for battery 2CD because it would have resulted in the battery not being able to meet it's composite duty cycle. The licensee documented the degraded 2CD battery in CR 00-01852 and was tracking battery capacity. The licensee was evaluating methods to restore acceptable margin to the battery on a post restart schedule.

The acceptance criteria for device voltages is that the voltage at the device must be greater than the minimum rated operating voltage plus one volt. The system analysis calculations (2-E-N-ELCP-250-006 and 2-E-N-ELCP-250-007) for the AB and CD batteries each listed in Attachment C of the calculations, a number of devices that fail to meet this criteria. Attachment P of the calculations documented the licensee's resolution of the devices identified in Attachment C. The licensee analyzed each device and concluded that the low voltage was acceptable in each case or that the device was not required to function when the voltage is low. Therefore, the licensee concluded no modifications were necessary. The team considered the licensee's position provided an adequate technical resolution for this issue.

The licensee had previously identified in CR 00-00450 that the stated acceptance criteria of calculation 2-E-N-ELCP-250-001 had not been met and recommended that the calculation be revised post restart to clearly demonstrate compliance with the plant's licensing basis. This calculation demonstrated (based upon new fuses being installed under DCP-4392) that coordination exists between upstream and downstream devices except for two cases. One case involved an emergency Appendix R cable that is

normally not connected, but can be connected to supply the Unit 1 power operated relief valves. The second case involved power supplies to the balance of plant (BOP) valve panels. Additionally, this calculation demonstrated that protection of cables is provided except for seven of twenty two cases examined. The calculation justified that the seven cases were satisfactory based upon the unprotected cables being BOP cables or run in separate cable trays and thus met single failure criterion. The team considered that this calculation provided an adequate technical resolution to these issues.

c. Conclusions on Electrical Protection Coordination Program Ready for Restart

The team considered the Electrical Coordination program assessments and corrective actions comprehensive and that they encompassed appropriate facets of the plants electrical design. The corrective actions included plant modifications and calculations that demonstrated appropriate electrical protection and coordination in the plant electrical systems. However, the team identified a weakness associated with the lack of design basis documentation available to address the 4KV motor cable protection against short-circuits. Based on a review of the completed major corrective actions for Unit 2 and the planned corrective actions for Unit 1, the team concluded that an adequate basis for electrical plant coordination and protection would be established. Therefore, CSC Item No. 14E and RAM Item No. R.2.14.1 are closed.

A non-cited violation was issued for the licensee's failure to perform an adequate design review and identify the correct relay function in an electrical containment penetration protection calculation.

E8.4 (Open) Emergency Diesel Generator Air System (RAM Item No. R.1.28)

a. Inspection Scope

The licensee identified in LER 50-315/99011-00 that the air system for the emergency diesel generators (EDG) might not support long term operability due to an error in the original system design. This issue was incorporated into the NRC RAM and classified as high priority inspection item R.1.28. The team reviewed corrective actions documented in the Engineering Action Plan 99-282, "EDG Starting Air," CR 99-03087 and CR 99-03167 and performed a walk down of the system.

b. Observations and Findings

The air system of the EDG provided the function of EDG starting, instrument air supply for selected instruments and EDG shutdown via the throttle control cylinder. The system included two safety grade receivers capable of two EDG starts from each air receiver and air for the instrument air system. Make-up air to the receivers was supplied by two non-safety/non-seismic systems. The air supply pressurized the receivers and provided make-up air following the EDG start to enable the EDGs to continue to run. Because makeup air is needed from the non-safety portion of the air system, long term operation of the EDGs could not be demonstrated. The licensee reported this condition to the NRC in LER 50-315/99011-00.

b.1 EDG Air System Corrective Actions

Engineering action plan 99-282 described the problems and resolutions for addressing the air system long term operability issue. Corrective actions to address the problems included design changes for the Unit 2 EDG air systems. However, the team identified that action plan 99-282 only described corrective actions for Unit 2. The licensee stated that this was an oversight and intended to revise the plan to include similar corrective actions for Unit 1.

The EDG air start system is supplied by air which is not dehumidified. The team identified a potential concern for corrosion/degradation of the air system caused by moisture carry-over during previous system operation. The licensee documented their response to this concern in CR 99-11498. The licensee identified that the blowdowns of the starting air receiver tanks performed each shift removed oil and water which had accumulated. The licensee stated, "this minimizes any water evaporation and entrainment into the compressed air starting system." Additionally, the licensee had performed inspections of the interior surfaces of the starting air receivers. These inspections had identified blistering of the internal tank coating and pitting corrosion in the lower portion of the air receivers. The licensee had removed and recoated the affected areas of the air receivers and had replaced the sections of the piping from the compressors to the air receivers as part of DCP 487 and DCP 549. The licensee also identified a need to perform a post restart revision during the next Unit 2 refueling outage to the existing design change packages for the air system, to install an air dryer system. These actions appeared appropriate to address the issue of moisture intrusion into the EDG air start systems.

b.2 EDG Air System Modifications

The team reviewed 2-DCP-487, "2-AB EDG Safety Related Compressor/Piping," and 2-DCP-549, "2-CD EDG Safety Related Compressor/Piping." These modifications replaced the piping (including a check valve) from the air compressors to the receivers and the air compressors for the 2AB and 2CD EDG air systems. Additionally, the team performed a walkdown of the areas affected by these design changes (piping replacement was in progress). The team did not identify any deficiencies with the physical work proposed under these design changes. However, the licensee intended to demonstrate system operability during the post-modification test, which had not been written at the time of this inspection. The licensee issued 2-FCN-549-R1-05 to document the proposed test plan. The team noted the leakage rate identified in this FCN was not consistent with that determined in calculation MD-12-DG-003-N, and the system acceptance criteria were not tied to a documented engineering evaluation which could be extrapolated to the operating conditions expected under accident conditions.

b.3 EDG Air System Calculations

The team reviewed MD-12-DG-003-N, "Capacity Requirement for Emergency Diesel Generators Starting Air System." In this calculation, the licensee determined the time required to recharge the air receiver after a design basis demand. However, no system performance criteria or acceptance criteria were provided in this calculation with respect to demonstrating long term system operation under accident conditions. Further, the

system leakage rate assumed in this calculation was much less than that assumed in the test plan outlined in 2-FCN-549-R1-05. Additionally, this calculation did not attempt to calculate the pressure drop across the newly installed system piping and check valve and instead the licensee appeared to be basing operability solely on the post modification testing of the system. Therefore, the team could not verify system operability pending review of the completed post modification tests.

c. Conclusions on EDG Air System Operability

The team could not reach a conclusion on the long-term operability of the EDG air systems. Design changes to these systems included installation of new air supply piping and air compressors. However, the design changes and supporting calculations relied on a post modification test to demonstrate system operability which had not been issued. Additionally, the action plan for resolution of this issue did not specify the actions to be taken to resolve the operability for Unit 1 EDG air systems. Pending review of the revised action plan and testing, which demonstrates operability for the EDG air systems, this issue will remain open.

E8.5 (Closed) Control of Fibrous Material Within Containment (RAM Item Nos. R.2.3.5 and R2.3.7)

a. Inspection Scope

The NRC had previously identified concerns associated with control of transportable debris that could affect the containment sump suction strainers and adversely affect the emergency core cooling systems. Specifically, the NRC identified in EEI 50-315/97017-01, EEI 50-316/97017-01, EEI 50-315/97017-04, and EEI 50-316/97017-04 that the licensee failed to establish sufficient measures to assure that the design basis was correctly translated into specifications and procedures related to the installation of fibrous material within containment. These issues were incorporated into the NRC RAM and classified as high priority inspection Items R.2.3.5 and R.2.3.7. The team reviewed corrective actions for this issue described in Engineering Action Plan CNT-99-269, which included recent revisions to 12-CHP-5021.ECD.005, “Installation, Replacement, and Repair of Silicone Fire Barrier Penetration Seals,” and specification, DCC-FP-101-QCN, “Fire Barrier Penetration Seals for Openings in Fire Rated Walls, Floors, or Ceilings.”

b. Observations and Findings

The actions specified by the licensee to resolve deficiencies associated with control of fibrous materials in containment were identified in Engineering Action Plan CNT-99-269. The principal actions included requirements to revise the specification and procedure for control of fibrous materials in containment. The team reviewed the latest revisions to specification DCC-FP-101-QCN, Revision 9, and 12-CHP-5021.ECD.005, Revision 14. The team noted that previous issues associated with these documents described in inspection report 50-315/99029(DRS); 50-316/99029(DRS) had been adequately addressed except for two editorial issues. The team identified a note in Section A.1 and A.2 of Appendix A to 12-CHP-5021.ECD.005, which contained potentially misleading provisions regarding whether it applied in containment or not. The licensee agreed with

the team that this was confusing and agreed to clarify this area in a revision to this document which was currently under preparation. Additionally, the team identified that a statement in Section 2.0 of DCC-FP-101-QCN, which stated, “Also fire stops in cable trays,” which was misleading. The licensee stated that this statement was intended to include fire stops within the full scope of the specification. The licensee agreed that this statement was not clear and committed to revise the wording in the next revision of the specification. Based on the completed actions and planned editorial corrections, the team considered that these documents adequately addressed the issues and RAM Items R.2.3.5 and R.2.3.7 are closed.

c. Conclusions on Control of Fibrous Material in Containment

The revised sump action plan, revised specification and procedure controlling fibrous materials in containment were adequate to resolve previously identified concerns in these areas and RAM Items R.2.3.5 and R.2.3.7 are closed.

E8.6 High Energy Line Break Issues (RAM Item Nos. R.1.14, R.1.35, and R.1.24)

a. Inspection Scope

The licensee identified deficiencies in the High Energy Line Break (HELB) Program that resulted in unanalyzed conditions for safety related systems. These conditions were reported to the NRC in LER 50-316/98005-00 on the CCW system, LER 50-315/98058-00 on the AFW system, and LER 50-315/99026-00 on the overall HELB program. These LERs reported conditions were incorporated into the NRC RAM and were classified as high priority inspection Item Nos. R.1.14, R.1.24, and R.1.35, respectively.

The team reviewed licensee root causes, HELB action plan, a sample of the licensee’s HELB design changes and calculations related to HELB upgrades to the plant, the licensee’s walkdown plan, HELB area drawings, target lists, evaluations, and other HELB related corrective actions. The team also observed the in-plant locations for the proposed plant design changes to address concerns for HELB affects on the CCW system, AFW system, and plant electrical systems.

b. Observations and Findings

On October 22, 1999, the licensee determined that, based on the accumulation of numerous HELB related condition reports, some locations in the plant could be considered unprotected from the effects of a postulated nearby HELB event. The safety related equipment at these plant locations was either not qualified for the harsh environment that would result from a HELB, or might have been damaged by the jet impingement from a crack in the nearby high energy piping. The plant locations and equipment affected by this condition were documented in LER 50-315/99026-00. The team reviewed the licensee root cause evaluation for these conditions and corrective actions as described in the following sections.

b.1 HELB Root Cause Assessment

During the ESRR program, the licensee identified several HELB-related deficiencies. The root cause and corrective actions for the HELB program issues was documented in CR 99-11773. The root cause was documented as “organizational and programmatic breakdown of the design control process” that resulted in “a lack of clear definitions and incomplete calculations.” The licensee identified that “most of the errors were made,” during the original FSAR, Appendix O, submittal. Contributing to the problem was the fact that significant NRC HELB-related guidance was developed after the plant was near completion, making any corrective actions more difficult and beyond initial plant license requirements. Further, the HELB and post-Three Mile Island related environment qualification programs were not coordinated, and HELB problems were not adequately identified, evaluated, and dispositioned.

The team reviewed licensee identified HELB related deficiencies and attended the Corrective Action Review Board (CARB) meeting in which the licensee formally reviewed and accepted the “Root Cause Analysis for Condition Report CR 99-11773 Concerning High Energy Line Break (HELB) Issues.” The CARB raised several questions on the proposed actions identified for the root causes. These questions prompted changes to the HELB program, such as upgrades in the training program to incorporate the current industry practice and operating experience on HELB issues. Based on this observed meeting, the identified HELB related deficiencies and review of the Appendix O, of the FSAR, the team considered that appropriate root causes and corrective actions (as discussed in the following sections) had been implemented for the HELB program issues.

b.2 HELB Program Reconstitution

As a corrective action for the HELB program issues, the licensee assigned a dedicated resource for program ownership within the organization and committed to upgrade the HELB design basis documentation and related plant procedural processes. The licensee identified several procedures that needed to be upgraded for HELB coordination (action 17 of CR 99-11773). The team considered that the procedure upgrade program, coupled with a dedicated HELB program owner, was an appropriate course of action for resolving these HELB issues.

The licensee contractor studied the HELB issues, developed walkdown and upgrade plans, identified HELB targets, dispositioned those targets, and produced the necessary documentation to reconstitute the licensee’s HELB program. These actions also fulfilled the licensee’s Engineering Leadership Plan, Problem Statement 6, Action Item 4.3 requirement to perform a detailed assessment of the HELB program. An additional HELB program self-assessment was planned by the licensee, to be performed approximately one year following restart (tracking number SA-2001-NED-001).

The results of the initial contractor assessment of the HELB program, which began in 1998, were provided to the licensee in a June 30, 1999 report, “HELB Phase 1 Final Report, Program Integration Study.” This report noted UFSAR and FSAR documentation deficiencies for systems inside containment, weaknesses in compartment pressure and temperature analyses, the lack of a verified HELB Safety

System Equipment List, and the potential for piping whip restraint deficiencies. This study identified that the original HELB program was burdened by overly conservative requirements, which resulted in an unwieldy program. The suggested corrective action was improvement in program efficiency by adopting less restrictive HELB requirements allowed under current regulatory requirements and industry practices. For example, this study recommended elimination of the requirement to specify arbitrary intermediate breaks and to establish a leak-before-break criterion.

The licensee's pre-restart HELB program upgrades included establishment of appropriate HELB design criteria, development of a project plan, documentation of the plant HELB licensing basis and commitments, and completion of preliminary calculations. The licensee contracted a vendor to provide a comprehensive review of industry practice and regulatory requirements. This was documented in the June 30, 1999 "HELB Phase 1 Final Report, Program Integration Project." Based on review of this document, the team determined that the licensee had developed appropriate HELB regulatory criteria and incorporated acceptable industry practices. The team noted that this document had not been subject to an OAR, since it predated this process.

The licensee's contractor conducted extensive safety system and safe shutdown system walkdowns to identify and document potential HELB vulnerabilities. The team reviewed the licensee's "HELB Walkdown Plan," a systematic set of HELB zone drawings that covered all plant areas containing high energy piping systems, the "HELB Program Walkdown Report" and the "HELB Program Target Evaluation Report," which documented the target disposition rationale. The team noted that these documents had not been subject to an OAR, since it predated this process. The Director of Design Engineering committed to the team that the final documents used as a basis for the HELB program would undergo an OAR.

The "HELB Program Target Evaluation Report" documented a systematic and comprehensive as-built inspection and survey of plant areas and rooms to identify and evaluate safety-related targets susceptible to HELB. More than a thousand potential targets were identified and screened, resulting in additional technical evaluation of more than 40 targets. The final HELB walkdown scope focused on plant vulnerabilities from steam generator blowdown system piping, chemical and volume control system piping, main steam lines to the turbine driven AFW pumps, and the AFW lines from the AFW system discharge check valves to the main feed system piping. The walkdown scope included piping and the associated supports inside and outside containment for piping larger than one inch diameter. The purpose of the walkdowns was to verify UFSAR information and to collect HELB data. The team considered this report comprehensive.

The reconstitution effort included HELB reviews, verifications, and upgrades intended to bring the physical plant in line with industry practice and into conformance with regulatory guidelines. Overall, the team considered that the resulting documentation reflected a significant level of effort to comprehensively address reconstitution of the plant's HELB design basis.

b.3 Pressurizer Surge Line Whip Restraint Modification

The team inspected the key physical plant areas impacted by the HELB design change packages and observed HELB related design changes in progress. While design basis testing for HELB related design changes was not practical, the team questioned the licensee's basis for not performing a post modification check of the pressurizer surge line pipe whip restraint gaps.

The licensee performed an updated pressurizer surge line pipe whip restraint calculation (draft calculation SD-990825-001) based on dynamic methods rather than the previous static methods. Based on this calculation, the licensee identified several pressurizer surge line whip restraints with excessive gaps which required shimming to mitigate the affects of a postulated HELB event. The installation of these shims was to be implemented in 2-DCP-4260, "Modification to Surge Lines Pipe Whip Restraints." However, the team identified that no post modification checks were planned to verify that adequate clearance (e.g., gaps) between the shimmed whip restraint and the surge line would exist under hot plant conditions. The licensee was dependent on the accuracy of vendor provided data on thermal expansion growth, which indicated that gaps would exist under hot plant conditions. The licensee staff could not provide the vendor calculations which supported the predicted piping growth and estimated hot plant gaps. The team noted that an error in the predicted hot plant gaps or modification installation could result in the surge line contacting a whip restraint, creating unanalyzed loads on the surge line. Specifically, errors could arise from the differences in the as-built plant configuration (likely considering the complex shapes involved) as well as errors in temperature gradients assumed in the vendor thermal growth predictions. These error sources increased the likelihood that the predicted results were incorrect. Therefore, the team considered this issue an Inspection Followup Item (IFI 50-316/2000007-03) pending licensee resolution of the post modification gap checks for 2-DCP-4260.

b.4 Resolution of HELB for the Auxiliary Feedwater System

The licensee documented in LER 315/98-058-00 that the east motor driven auxiliary feed pump (MDAFP), AFW power supply cables to four AFW valves, and the turbine driven auxiliary feedwater pump (TDAFP) were susceptible to a HELB in the steam supply line. The licensee determined that a break in the TDAFP supply line could cause ambient temperatures to reach 330 degrees Fahrenheit, impacting all of these components through a shared hallway. Subsequently, during the ESRR, the licensee also determined that the AFW pumps could be adversely impacted by a postulated HELB event in the turbine building through the turbine building door and the ventilation system for the MDAFP room. Moreover, the licensee determined that AFW pump room fire dampers would not mitigate these conditions and might not work against a differential pressure (e.g., caused by steam flow).

The licensee analyzed these conditions and initiated physical plant upgrades to correct these vulnerabilities. The licensee proposed sealing the AFW pump rooms to ensure that the rooms did not communicate with the turbine building or each other. Sealing these rooms resulted in the need for the licensee to modify the ventilation systems for the AFW pump rooms. The proposed modification incorporated an essential service

water cooled refrigeration-cycle room cooler to ensure that the AFW system would function under accident conditions. The proposed modifications involved an unreviewed safety question in accordance with 10 CFR 50.59, since the probability of malfunction of the new cooling systems for the AFW pump rooms are higher than those for the current ventilation system. The licensee submitted a “License Amendment Request Modifications To Auxiliary Feedwater Pump Room Cooling” dated February 18, 2000, to document this unreviewed safety question. The final design changes had not been issued at the time of inspection. Preparation for installation of the AFW pump room cooler was underway using an “at risk” (e.g., without supporting calculations or a safety evaluation) design change package 2-DCP-4261a “Unit 2 Auxiliary Feedwater Pump Room Ventilation.” The team reviewed the at risk design change package, the related drawings, and determined these changes appeared adequate to resolve this HELB vulnerability.

b.5 Resolution of HELB for Safety and Non-Safety Related Switchgear

During the ESRR the licensee identified safety and non-safety related switchgear equipment potentially subject to HELB harsh environmental effects that could be created by a break or crack in the feed water heaters or the associated piping systems in the turbine building. Specifically, the licensee questioned the adequacy of UFSAR statements that claimed HELB protection for the switchgear room equipment was provided by roll up doors and fire protection equipment. The affected equipment included safety related 600VAC and 4160VAC switchgear, control room instrument distribution inverters, and the AB battery train.

The licensee planned to initiate design change 2-DCP-4247, “Installation of Exhaust Flow Paths in the Unit 2 4KV Switchgear Area,” and RPA-4535, “4KV SWGR [switchgear] Area Jet Impingement Shield,” to provide back-draft dampers and a jet impingement shield to address the concerns for HELB on switchgear room equipment. These modifications had not yet been approved. The team concluded that based on review of the draft 2-DCP-4247 and changes proposed under RPA-4535 that the planned corrective actions should correct this HELB vulnerability.

b.6 Resolution of HELB for Emergency Diesel Generators (EDG)

During the ESRR, the licensee identified that the EDG rooms might be adversely impacted by a HELB event. The EDG room exhaust air ventilation duct to the turbine building includes fire dampers, but these dampers would not close as a result of a turbine building HELB event. Thus, a postulated HELB event in the turbine building could create a harsh environment in the EDG room and the EDG room equipment was not designed for a harsh environment. Based on the results of calculation TH 96-01, the licensee determined that a main steam line break could result in the loss of an EDG.

Licensee corrective actions for the EDG HELB issue included installation of back draft dampers on the four EDG room exhaust ventilation systems under 12-DCP-4254, "Add Back Draft Damper to EDG Vent System." This DCP had not been issued yet and thus was not available for review. Based on the existing evaluations and the scope of the proposed design change, the team concluded that the changes should adequately address this HELB vulnerability.

b.7 Resolution of HELB for Component Cooling Water System

The licensee identified steam and feedwater piping, located in a pipe chase adjacent to the Unit 2 CCW system pumps, which could be accessed through three adjacent doors. This configuration represented a HELB vulnerability for the Unit 2 CCW pumps, since a postulated steam or main feed line break from adjacent lines could open the access doors and create a harsh environment for the CCW pumps. The CCW pump motors and other equipment were not qualified for the associated harsh environment expected, if these doors failed due to a HELB. These conditions are documented in LER 315/98-005-00 and CR 98-2383.

The licensee identified mitigating conditions for this HELB concern, such as the pump area being open to the rest of the auxiliary building and the fact that the CCW pump motors are provided with a ventilation system considered adequate to protect the pump motors from the postulated harsh environment. Nevertheless, the licensee was proceeding with a design change to reinforce the access doors on the pipe chase to ensure that these doors would not fail open under worst case loading attributed to a postulated critical crack in the nearby main steam piping. The team reviewed the design change package 2-DCP-4258, "Structural Door Restraints for Nos. 406, 407, 408," and observed the in-plant status of this work. This modification installed a metal strong-back across the access doors to the main steam and feedwater piping chase and appeared adequate to resolve this HELB vulnerability.

b.8 Resolution of HELB for the Turbine Driven Auxiliary Feed Pump (TDAFP) Battery

The 250 VDC N–train battery and related equipment provide power needed by the TDAFP. In Unit 2, the N-train battery support equipment located outside the battery room was found during the ESRR to be susceptible to HELB effects. Specifically, a postulated HELB in the adjacent SG blowdown piping could create a harsh environment for the battery.

To resolve this concern, the licensee proposed to use the NUREG-0800, "Standard Review Plan," Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," pipe stress analysis provision for postulated crack locations, specifically for a relatively small portion of steam generator blowdown piping in the general vicinity of the 250 VDC N-train batteries. To meet the pipe stress analysis provision, the licensee was installing blowdown pipe under 2-DCP-4253, "Protection of N-Train Battery - EL. 633' " with over-design pipe wall thickness. The team observed this work in progress and concluded that this modification would adequately resolve this HELB vulnerability.

b.9 Resolution of HELB for the Cables and Conduits Inside Containment

The licensee identified more than forty cable and conduit jet impingement targets inside containment that required more detailed review to assess whether the targets needed to be protected. The licensee evaluated these potential targets and determined that HELB protection was not required. This determination was based on more detailed piping stress analysis (Generic Letter 87-11), a determination that targets were outside the range of the postulated jet, or a determination that the target was not required for safe

shutdown. The team considered the licensee's technical approach to target evaluations appropriate to resolve this issue.

c. Conclusions on HELB Issues

The licensee had discovered deficiencies in the methodology used to resolve previous HELB issues, HELB related licensing commitments, and HELB associated program documentation. The team determined that the overall level of effort being applied to upgrade the licensee's HELB program was substantial and consistent with that needed to resolve the deficiencies. The licensee's upgrade program included reanalysis, revised program documentation, and physical plant changes. Overall, the corrective actions implemented by the licensee adequately addressed the HELB related deficiencies and RAM Item Nos. R.1.14, R.1.24, and R.1.35 are closed.

The team questioned the licensee's basis for not performing a post modification hot plant check of the pressurizer surge line pipe whip restraint gaps and considered this issue an Inspection Followup Item.

E8.7 (Open) Refueling Water Storage Tank Level Instrument Uncertainty (RAM Item No. R.2.3.2)

a. Inspection Scope

The licensee identified in LER 50-315/97011-02 the potential for the refueling water storage tank (RWST) level instrumentation uncertainties to lead to premature swap over of emergency core cooling systems to recirculation sump operation. This condition was reported to the NRC in LER 50-315/97011-02. This issue was incorporated into the NRC RAM and classified as a high priority inspection Item R.2.3.2. This issue had been previously reviewed during NRC inspection 50-315/99029(DRS); 50-316/99029(DRS) and could not be closed due to the lack of issued supporting calculations. This inspection focused on calculations and testing used by the licensee to determine level uncertainties.

b. Observations and Findings

During a previous NRC inspection, (Report 50-315/99029(DRS); 50-316/99029(DRS)) the NRC had identified that neither calculation 1-2-UNC-339 CALC1, "Refueling Water Storage Tank Level Loop Accuracy Calculation," Revision 1, nor 1-2-UNC-339 CALC2, "Setpoint Calculation for RWST Level Alarms and RHR Pump Trip Interlock," draft Revision 0, dated September 29, 1999, included the effect of potential RWST vacuum during the injection phase. The team determined that the potential vacuum affects as determined in MD-2-RWST-001-N had been adequately considered in 1-2-UNC-339 CALC2, "Setpoint Calculation for RWST Level Alarms and RHR Pump Trip Interlock," Revision 1. However, the appropriate level uncertainty assumed at the lowest indicated level allowed in the RWST tank (7 percent) for EOP actions was to be confirmed by testing. This testing had not yet been completed and the licensee intended to use the testing results to confirm that tank vortexing (which could introduce air into the emergency core cooling system (ECCS) pumps) would not occur at the lowest usable tank volume.

The team reviewed a draft copy of ICP-00463 in which the licensee documented that information contained in DIT -B-00846-00 transmitting the results of RWST vortex testing validated the results of the uncertainty calculation 1-2-UNC-339 CALC2. This testing had not been completed and the team identified inconsistencies in the flowrate assumed from the tank. In DIT B-00459 the licensee selected a flowrate of 2000 gallons per minute (g.p.m.) as the appropriate flowrate during transition from injection to recirculation phase for ECCS. Scale model flow testing was being performed to confirm acceptability of the seven percent tank level point and 1800 g.p.m. flowrate. The licensee stated that the basis for the 2000 g.p.m. and the 1800 g.p.m. number were different calculations combined with engineering judgment. The team could not verify the licensee's conclusion that the1800 g.p.m. flowrate was a bounding number for the onset of air ingestion at an indicated seven percent tank level because of a lack of a documented basis for this number. Therefore, this issue will remain open pending demonstration that the 1800 g.p.m. flowrate was appropriate to ensure air would not be introduced into ECCS pumps at the seven percent indicated RWST level.

The team reviewed calculation MD-2-RWST-001-N, "Maximum Differential Pressure for RWST Vent Path," and identified an administrative problem with the calculation. The calculation was issued in an "unrestricted" status, which meant that there were no limitations or outstanding data to verify. However, the team identified that the acceptance criterion of this calculation were obtained from a restricted calculation SD-990914-002, "Structural Evaluation of RWST Tank Water Level Increase." The licensee generated CR 00-03215 to document this error. The margin to the acceptance criteria for this calculation was sufficient for the team to determine that this issue would not affect the conclusions of the calculation.

c. Conclusions on RWST Level Uncertainty

The licensee had completed many of the corrective actions associated with establishing uncertainty for the RWST level. However, the testing to confirm that RWST vortexing would not occur at the lowest useable tank volume had not been completed. Additionally, the team identified a lack of a documented basis for the flowrate used in confirming the absence of vortexing at the lowest useable RWST volume. Pending review of the testing and analysis to confirm the lowest useable RWST volume, this issue remains open.

E8.8 Containment Level Uncertainty (RAM Item No. R.2.3.25)

a. Inspection Scope

Accurate containment level instrumentation is needed for operators to verify the availability of sufficient water in containment to initiate the sump recirculation operations to mitigate the consequences of a LOCA. The NRC identified in EEI 315/316/98009-005 that the indicated post-accident containment water level could not be accurately determined due to the licensee's failure to consider level indication uncertainty and vortexing, air entrainment, or net positive suction head requirements in supporting calculations for the containment sump level instrumentation loops. This issue was incorporated into the NRC RAM and classified as a high priority inspection Item, R.2.3.25.

The team reviewed licensee corrective actions documented in CR 97-2312 and CR 97-2350 for this issue.

b. Observations and Findings

The licensee's corrective actions for this issue included an analysis of containment water inventories. The team reviewed DIT-B-00187-01 dated December 22, 1999. This DIT documented the licensee's analysis, which included small and large pipe breaks under various operating conditions for the ECCS and the containment spray system. The team noted that the analysis performed reflected a detailed understanding of the containment water level uncertainty issue.

The key corrective action for this issue included installation of more accurate containment sump level indication instrumentation. The team reviewed 2-DCP-4346, "Install New Containment Water Level Switches," and supporting calculations related to the installation of four containment level indication float switches. This DCP described the installation of two trains of high and low level switches which would allow the operators to more accurately determine containment water level conditions, to ensure that sump recirculation operations would not be started prior to attaining an adequate inventory level in the sump. 2-DCP-4346 included discussions of electrical separation, power supply loading, and fusing that indicated a comprehensive review of potential problems. The supporting calculation SD-991203-013, "Qualification for Conduit Supports and Mounting Details of Flood Level Switches 2-NLI-330, 2-NLI-331, 2-NLI-340, and 2-NLI-341," documented a detailed equipment qualification, including seismic analyses, for the supports and mounts for the conduits installed under this DCP. The team considered that this design change provided an appropriate technical solution for this issue.

Documentation provided in DIT-B-00187-01 appropriately supported 2-DCP-4346. Specifically, this DIT documented the float switch uncertainty limit of plus or minus 1.25 inches and described the basis for the low level switch installation point at sump elevation 602 feet, 2-3/4 inches. The level switches described in DCP 4346 were environmentally qualified, class1E, float type level switches which provided an actuation accuracy of plus or minus 1/4 inch. Because the proposed level circuits were more accurate and within the acceptance limit for uncertainty (plus or minus 1.25 inches), the original uncertainty questions associated with the previous level indication circuits were no longer applicable. Therefore, the team concluded that the level instrument modifications, when completed, would be adequate to support sump recirculation operations during emergency operations.

c. Conclusions on Containment Level Uncertainty

The team determined that corrective actions to install dedicated containment level instrumentation were adequate to resolve concerns for sump recirculation level uncertainty and RAM Item No. R.2.3.25 is closed.

E8.9 Low Priority Restart Issues

a. Inspection Scope

The team examined relevant inspection reports, LERs, CRs, action requests, and status reports to ensure that: the issue was captured in the licensee's corrective action system; the issue was correctly characterized and classified; appropriate corrective actions were specified; and the corrective actions were either completed or scheduled and tracked for completion.

b. Observations and Findings

For each of the issues listed below the team confirmed that the four attributes listed in the scope above were satisfied and therefore are closed.

- R2.3.21 URI 50-315/316/98007-15, "Upward Travel of the Blocks Would be Limited by Contact with the Loose Nuts"
- R2.3.39 URI 50-315/316/98009-24, "Apparent Failure to Maintain Proper Design Control Regarding Industry Standards and Codes"
- R2.7.1 EEI 50-315/98007-02, "An Apparent Violation for Declaring the Recombiner Operable with Recorded Data Which Exceeded the TS Limits"
- R2.7.2 EEI 50-315/98007-03, "An Apparent Violation for an Inadequate Procedure Which Failed to Measure Resistance to Ground Immediately Following the Heat Up Test"
- R2.3.48 LER 98013-01, "Improper Splice Configurations for Power Operated Relief Valve Limit Switches Results in Unanalyzed Condition"
- R.1.10 URI 50-315/97025-03, “SSPS Relay Operability History”
- R.2.4.9 LER 50-315/98002-01, “Degraded Solid State Protection System Master Relays Result in Condition Outside the Design Basis”

For each of the issues listed below the team could not confirm that the four attributes listed in the scope above would be satisfied and will remain open as discussed below.

- R.2.7.5 LER 50-315/98016-02, “Non-Safety Related Cables Routed to Safety Related Equipment.” This issue will remain open pending completion of NRC internal reviews on the plant’s licensing basis in this area.
- R.1.32 LER 50-315/99022-00, “Electrical Bus Degraded Voltage Setpoints Too Low For Safety Related Loads.” On February 22, 2000, the licensee informed the team that LER 50-315/99022-00 was undergoing major revision to address the change of degraded voltage setpoints. This issue will remain open pending review of licensee's revised LER and corrective action plans for this issue.

c. Conclusions on Low Priority Restart Issues

Based on verification that the issues were entered in the corrective action system (the issues were properly characterized and classified, appropriate corrective actions had been specified, and the corrective actions were scheduled and tracked) seven of the nine low inspection priority issues identified above are closed.

X. Management Meetings

X1 Exit Meeting Summary

The team discussed the progress of the inspection with licensee representatives on a daily basis and presented inspection results to members of licensee management at the conclusion of the inspection on March 15, 2000. The licensee acknowledged the findings presented and did not identify any of the potential report input as proprietary.

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INSPECTION PROCEDURES USED

IP 37551:	Engineering
IP 37700:	Design Changes and Modifications
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 92903:	Follow-up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Report Item No.	Item Type	RAM Item No.	Description
50-316/2000007-01(DRS)	NCV	NA	Failure to perform an adequate design review for the SGTR margin to overfill analysis report (AEP 99-486).
50-316/2000007-02(DRS)	NCV	NA	Failure to perform an adequate design review of calculation 2-E-N-PROT-PEN-001/0 and identify the appropriate protective relay.
50-316/2000007-03(DRS)	IFI	NA	Resolve pressurizer surge line whip restraint gap verification.

Closed

Item No or Report Section No.	Item Type	RAM Item No.	Description
50-316/2000007-01(DRS)	NCV	NA	Failure to perform an adequate design review for the SGTR margin to overfill analysis report (AEP 99-486).
50-316/2000007-02(DRS)	NCV	NA	Failure to perform an adequate design review of calculation 2-E-N-PROT-PEN-001/0 and identify the appropriate protective relay.
Section E8.1	NA	CSC 3A	Inadequate Design Control Pertaining to Uncontrolled and/or Unintended Changes in the Plant Design
Section E8.1	NA	Encl(2) Item C.3.2.e	Effectiveness of corporate engineering support.
Section E8.1	NA	Encl(2) Item C.3.2.f	Effectiveness of corporate design modification process.

Item No or Report Section No.	Item Type	RAM Item No.	Description
Section E8.3	NA	CSC 14E	Electrical Protection Coordination including Fuse/Breaker Control Program Ready for Restart
50-316/99027-00	LER	R.2.14.1	Underrated Fuses Used in 250 VDC System Could Result in Lack of Protective Coordination
50-315/97017-01; 50-316/97017-01	EEI	R.2.3.5	The Lack of Sufficient Measures to Assure That the Design Basis Was Correctly Translated into Specifications to Control the Installation of Material That Could Be Essential to the Safety-Related Functions of the Containment System
50-315/97017-04; 50-316/97017-04	EEI	R.2.3.7	A Procedure for Installation, Replacement, and Repair of Silicone Fire Barrier Penetration Seals That Did Not Require That Fibrous Damming Material Be Removed or Encapsulated Following Sealing Operations Is Not Appropriate to the Circumstances
50-316/98005-00	LER	R.1.14	Potential for High Energy Line Break to Degrade Component Cooling Water System
50-315/99026-00	LER	R.1.35	High Energy Line Break Programmatic Inadequacies Result in Unanalyzed Conditions
50-315/98058-00	LER	R.1.24	Postulated High Energy Line Break Could Result in Condition Outside Design Basis for Auxiliary Feedwater
50-315/98009-05; 50-316/98009-05	EEI	R.2.3.25	The Uncertainty Calculations for the Containment and Containment Sump Level Instrumentation Loops Do Not Account for the Impact on the Post-Accident Containment Water Levels (calculations 1-2-N3-01, 1-RPC-14, and 2-RPC-14), and Do Not Consider the Potential for Vortexing, Air Entrainment, or Net Positive Suction Head Requirements

Item No or Report Section No.	Item Type	RAM Item No.	Description
50-315/98002-01	LER	R.2.4.9	Degraded Solid State Protection System Master Relays Result in Condition Outside the Design Basis
50-315/97025-03	URI	R.1.10	SSPS Relay Operability History
50-315/98013-01	LER	R2.3.48	Improper Splice Configurations for Power Operated Relief Valve Limit Switches Results in Unanalyzed Condition
50-315/98007-03	EEI	R2.7.2	An Apparent Violation for an Inadequate Procedure Which Failed to Measure Resistance to Ground Immediately Following the Heat Up Test
50-315/98007-02	EEI	R2.7.1	An Apparent Violation for Declaring the Recombiner Operable with Recorded Data Which Exceeded the TS Limits
50-315/98009-24; 50-316/98009-24	URI	R2.3.39	Apparent Failure to Maintain Proper Design Control Regarding Industry Standards and Codes
50-315/98007-15; 50-316/98007-15	URI	R2.3.21	Upward Travel of the Blocks Would be Limited by Contact with the Loose Nuts

Discussed

Item No. or Report Section No.	Item Type	RAM Item No.	Description
Section E8.2	NA	CSC 11	Resolution of Hydrogen Mitigation System Operability and Material Condition Issues
50-315/99011-00	LER	R.1.28	Air System for Emergency Diesel Generators (EDG) May Not Support Long Term Operability Due to Original Design Error
50-315/97011-02	LER	R.2.3.2	Operation Outside the Design Basis for ECCS and Containment Spray Pumps for Switch over to Recirculation Sump Suction”

Item No. or Report Section No.	Item Type	RAM Item No.	Description
50-315/98016-02	LER	R.2.7.5	Non-Safety Related Cables Routed to Safety Related Equipment
50-315/99022-00	LER	R.1.32	Electrical Bus Degraded Voltage Setpoints Too Low For Safety Related Loads

LIST OF ACRONYMS USED

A	Ampere
AFW	Auxiliary Feedwater
CARB	Corrective Action Review Board
CCW	Component Cooling Water
CEQ	Hydrogen Skimmer and Air Recirculation
CFR	Code of Federal Regulations
CNP	Cook Nuclear Plant
CR	Condition Report
DCP	Design Change Package
DIT	Design Information Transmittal
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESRR	Expanded System Readiness Review
ESW	Essential Service Water
FSAR	Final Safety Analysis Report
HELB	High Energy Line Break
IFI	Inspection Followup Item
IR	Inspection Report
KV	kilo-volt
LOOP	Loss of Off-site Power
MDAFP	Motor Driven Auxiliary Feedwater Pump
NCV	Non-Cited Violation
OAR	Owners Acceptance Review
RAM	Restart Action Matrix
RAP	Restart Action Package
RCP	Reactor Coolant Pump
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIDS	System Index Database System
SWGR	Switchgear
TDAFP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VDC	Volts Direct Current
VAC	Volts Alternating Current

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2-LDCP-0707, “Control Room DC Emergency Lighting”
2-LDCP-4369, “400A Fuses and Fuse Holder Block on Switchgear 2-21D3"
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2-DCP-549, “2-CD EDG Safety Related Compressor/Piping,” Revision 1
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1-2-EDS-166-2, Termination of Cable Shield at Switchgear, July 9, 1998
E-1000B-12, Unit 2 One Line Diagram, October 6, 1999

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1-2-UNC-339 Calc1, "Refueling Water Storage Tank Level Loop Accuracy Calculation," Revision 1
1-2-UNC-339 Calc 2, “Setpoint Calculation For RWST level Alarms, RHR Pump Trip Interlock, and Operation Points,” Revision 1
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