May 30, 2001

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

SUBJECT: D.C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 50-315/01-09(DRP); 50-316/01-09(DRP)

Dear Mr. Powers:

On May 12, 2001, the NRC completed an inspection at your D.C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on May 16, 2001, with Mr. Rencheck and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because the issue has been entered into your corrective action program, the violation is being treated as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the D.C. Cook facility.

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

Sincerely,

/RA/

Geoffrey E. Grant, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 50-315/01-09(DRP);
50-316/01-09(DRP)

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

DOCUMENT NAME: G:\COOK\dcc200109 drp.wpd

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

OFFICE	RIII	E	RIII	E	NRR		RIII			
NAME	DPassehl/trn		AVegel		JThompson		GGrant			
DATE	05/29/01		05/30/01		05/30/01		05/30/01			

OFFICIAL RECORD COPY

ADAMS Distribution:
CMC1
DFT
JFS2
GEG
HBC
BLB1
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/01-09(DRP); 50-316/01-09(DRP)
Licensee:	American Electric Power Company
Facility:	D.C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	April 1, 2001 through May 12, 2001
Inspectors:	B. L. Bartlett, Senior Resident Inspector D. Chyu, Region III Inspector K. A. Coyne, Resident Inspector J. D. Maynen, Resident Inspector R. A. Winter, Region III Inspector
Approved by:	A. Vegel, Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000315-01-09(DRP), IR 05000316-01-09(DRP); on 04/01-05/12/2001, Indiana Michigan Power Company, D.C. Cook Nuclear Power Plant, Units 1 and 2. Resident Inspector Report. Surveillance Testing.

This report covers a six-week routine inspection. The inspection was conducted by resident and Region III inspectors. One Green finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violations.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

GREEN. A non-cited violation was identified for the failure to ensure that the acceptance criteria contained in test procedures associated with the measurement of the reactor coolant seal injection line resistance adequately incorporated limitations associated with steam generator replacement and instrument uncertainty. Specifically, the licensee failed to identify that the requirements of Technical Specification 4.4.6.2.1.c were non-conservatively impacted by installation of replacement steam generators. Additionally, the test acceptance criteria did not adequately consider instrument uncertainty over the range of expected test conditions.

The inspectors evaluated the risk significance of this issue using the Significance Determination Process. Based on a review of recent test data, the inspectors determined that the impact of this failure was bounded by existing margin. Therefore, this issue did not result in inoperability of the controlled leakage charging flow path. Consequently, this issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. (Section 1R22)

B. Licensee Identified Findings

No findings of significance were identified.

Report Details

Summary of Plant Status:

Unit 1 began the inspection period at approximately 66 percent power. The licensee had reduced power in order to clean main feedwater pump condensers. On April 1, 2001, full power was achieved following the maintenance activities. The unit remained at full power throughout the rest of the inspection period.

Unit 2 began the inspection period at full power. On April 1, 2001, power was reduced to 57 percent in order to clean main feedwater pump condensers. On April 4, 2001, full power was achieved following the maintenance activities. The unit remained at full power throughout the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Unit 1 Component Cooling Water

a. Inspection Scope

The inspectors performed a complete mitigating system walkdown of the Unit 1 Component Cooling Water (CCW) System. The inspectors reviewed ongoing system maintenance, open job orders (JOs), and design issues for potential effects on the ability of the CCW system to perform its design functions. The inspectors ensured that the configuration of the CCW system was in accordance with applicable operating checklists. The inspectors also performed a complete system status check, which verified acceptable material condition of system components, availability of electrical power to system components and essential support systems, and that ancillary equipment or debris did not interfere with system performance. The Unit 1 CCW system was selected for this inspection based on its importance as a mitigating system used to prevent core damage. As part of this inspection, the inspectors reviewed the licensee's historical computerized issue tracking and job order data base.

b. Findings

No findings of significance were identified.

.2 Partial Equipment Alignment of Unit 1 East Motor Driven Auxiliary Feedwater Pump

a. Inspection Scope

The inspectors conducted a partial mitigating system walkdown of the Unit 1 East Motor Driven Auxiliary Feedwater Pump (MDAFWP) while the Unit 1 West MDAFWP was out of service for a surveillance test. The AFW system was selected for this inspection based on its importance as a mitigating system used to prevent core damage.

b. Findings

No findings of significance were identified.

.3 Partial Equipment Alignment of Unit 2 Control Air

a. Inspection Scope

The inspectors conducted a partial initiating events system walkdown of the Unit 2 Control Air System to verify that operation of the system was consistent with the Technical Specifications and licensing basis. The control air system was selected for this inspection based on its importance in causing a plant event if the system were to fail.

b. Findings

No findings of significance were identified.

.4 Unit 1 Control Room Air Conditioning

a. Inspection Scope

The inspectors conducted a partial system walkdown of the Unit 1 Control Room Air Conditioning system to verify that operation of the system was consistent with the Technical Specifications and licensing basis.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Zone Tours

a. Inspection Scope

The inspectors performed fire protection walkdowns of the following five risk-significant plant areas: the west end auxiliary building 633' elevation (Fire Zone 51), the auxiliary building 650' elevation (Fire Zone 69), the Technical Support Center (Fire Zone 126), the screen house (Fire Zone 142), and the fire pump house. The inspectors verified that fire zone conditions were consistent with assumptions in the licensee's fire hazard analysis. The inspectors walked down fire detection and suppression equipment, assessed the material condition of fire control equipment, and evaluated the control of transient combustible materials.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

On April 18, 2001, the inspectors observed a licensee fire drill which simulated a fire in the fire pump house. The inspectors evaluated the readiness of the licensee's personnel to prevent and fight this fire. The inspectors also attended the post-drill critique with the fire brigade.

b. Findings

No findings of significance were identified

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On May 10, 2001, the inspectors observed re-qualification training of licensed reactor operators, senior reactor operators, and non-licensed operators. The operators performed training on 10 CFR 50 Appendix R safe shutdown scenarios in the Unit 2 control room. The inspectors assessed communications and implementation of licensee emergency procedures.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

The inspectors evaluated the licensee's implementation of the Maintenance Rule (10 CFR Part 50.65), for three systems: containment isolation valves (CIV), CCW, and chemical and volume control. The inspectors assessed: (1) functional scoping in accordance with 10 CFR 50.65; (2) characterization of system functional failures; (3) safety significance classification; (4) 10 CFR 50.65 (a)(1) or (a)(2) classification for system functions; and (5) performance criteria for systems classified as (a)(2) or goals and corrective actions for systems classified as (a)(1).

.1 Containment Isolation Valve System

a. Inspection Scope

The inspectors reviewed the implementation of Maintenance Rule requirements for the containment isolation valve (CIV) system on both Units. The CIV system consists of all valves and system boundaries tested in accordance with 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

The Maintenance Rule function of the CIV system includes process line isolation in support of establishing containment integrity. The CIV Maintenance Rule system includes components from several systems, including: non-essential service water, component cooling water, sampling, ventilation, and emergency core cooling. The licensee recently completed historical reviews of CIV system performance and concluded that the system performance monitoring under the requirements of 10 CFR 50.65 paragraph (a)(2) was appropriate. The inspectors reviewed the results of the Maintenance Rule failure evaluations, the basis for system performance criteria, and discussed system performance and monitoring with engineering personnel. Because the CIV system Maintenance Rule function was associated with the containment fission product release barrier, the inspectors determined that this system was within the barrier integrity cornerstone.

b. Findings

No findings of significance were identified. However, during a review of historical CIV system performance, the inspectors identified that the licensee did not perform adequate Maintenance Rule evaluations of eight 10 CFR 50 Appendix J test failures that occurred since July 2000. The failure to appropriately evaluate the Maintenance Rule impact of Appendix J test results was associated with weak implementation of corrective actions for previously identified Maintenance Rule violations. This issue is discussed in additional detail in Section 4OA2, "Identification and Resolution of Problems," below.

.2 Chemical and Volume Control System

a. Inspection Scope

The inspectors reviewed the licensee's implementation of Maintenance Rule requirements for the Unit 2 Chemical and Volume Control System (CVCS). The licensee previously identified several repetitive maintenance preventable functional failures of Maintenance Rule functions, including: boric acid crystallization in the boration flowpath, CVCS unit cross connect valve leakby, and failure of CVCS throttle valves. Consequently, three CVCS Maintenance Rule functions were designated for monitoring under 10 CFR 50.65 paragraph (a)(1). The inspectors reviewed the CVCS goals and corrective actions, discussed system performance with the system manager, and reviewed recently completed functional failure evaluations. Because the CVCS provided reactor coolant system (RCS) inventory and reactivity control, the inspectors determined that Maintenance Rule implementation on the CVCS is predominantly associated with the mitigating systems cornerstone.

b. Findings

No findings of significance were identified. However, during a review of recent chemical and volume control system performance issues, the inspectors identified three examples of inadequate Maintenance Rule failure evaluations and one failure to properly account for equipment unavailability. Because these issues are related to the effectiveness of the licensee's corrective actions for previous Maintenance Rule violations, this issue is discussed in additional detail in Section 4OA2, "Identification and Resolution of Problems," below.

.3 <u>Component Cooling Water Train Isolation Valves (Both Units)</u>

a. <u>Inspection Scope</u>

As part of Engineering Action Plan 01-614, the licensee performed historical reviews of structures, systems, and components scoped into the Maintenance Rule to determine past system performance. The licensee documented the historical review of the component cooling water (CCW) system in Condition Report (CR) 00356032. The inspectors compared the licensee's results to a review of the job orders, CRs, and log entries regarding both units' CCW cross train isolation valves. These valves were scoped into the Maintenance Rule due to their function to isolate the "A" Train and "B" Train CCW systems from each other. The inspectors verified that the licensee had appropriately captured past reliability issues with the CCW cross train isolation valves.

b. <u>Findings</u>

No findings of significance were identified.

1R13 <u>Maintenance and Emergent Work</u> (71111.13)

.1 <u>Unit 2 West Essential Service Water Pump Outage</u>

a. <u>Inspection Scope</u>

On April 25, 2001, the licensee removed the Unit 2 "B" Train essential service water (ESW) pump from service for routine maintenance activities. In addition to reviewing the associated maintenance risk assessment, the inspectors walked down portions of the auxiliary feedwater system, essential service water system, emergency power systems, and auxiliary building general areas to verify that risk analysis assumptions were valid. The inspectors also verified that TS and Administrative Technical Requirements (ATR) were met during the time the ESW pump was inoperable. The inspectors discussed the risk control management with the shift technical advisor and the work control center senior reactor operator. The inspectors determined that the ESW pump outage impacted the mitigating systems cornerstone due to the loss of redundancy for post accident heat removal.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Maintenance During the Week of April 15, 2001</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the applicable maintenance job orders and clearance requests, reviewed the on-line maintenance risk evaluation, and assessed other equipment out of service that may have impacted the risk assessment of the work week schedule.

b. Findings

No findings of significance were identified.

.3 Maintenance During the Week of April 29, 2001

a. Inspection Scope

The inspectors reviewed the applicable maintenance job orders and clearance requests, reviewed the on-line maintenance risk evaluation, and assessed other equipment out of service that may have impacted the risk assessment of the work week schedule.

b. Findings

No findings of significance were identified.

.4 Unit 1 Turbine Driven Auxiliary Feedwater Pump

a. Inspection Scope

On May 10, 2001, operators removed the Unit 1 turbine driven auxiliary feedwater pump (TDAFWP) from service for routine maintenance. Removal of the pump from service resulted in an entry into the TS 3.7.1.2 Limiting Condition for Operation (LCO) because the maintenance required the TDAFWP to be made inoperable. Because the auxiliary feedwater (AFW) system provides the ultimate heat sink during certain postulated accidents, the inspectors considered this inspection to be part of the mitigating systems cornerstone. The inspectors reviewed the clearance order, job order, and control room logs for this planned LCO entry. In addition, the inspectors walked down portions of the motor driven AFW pumps to verify that the redundant trains were properly lined up.

b. Findings

No findings of significance were identified.

.5 Unit 2 “B” Train Emergency Diesel Generator Maintenance Outage

a. Inspection Scope

On May 3, 2001, operators removed the Unit 2 “B” Train emergency diesel generator (D/G) from service for routine preventive maintenance and minor corrective maintenance. Removal of the D/G from service resulted in an entry into the TS 3.8.1 Action Statement because the maintenance required the D/G to be made inoperable. Because the D/G’s provide the backup electrical power supply during certain postulated accidents, the inspectors considered this inspection to be part of the mitigating systems cornerstone. The inspectors reviewed the clearance order, job order, and control room logs for this planned TS entry. In addition, the inspectors walked down portions of the “A” Train emergency diesel generator and support systems to verify that the redundant trains were properly lined up.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Excessive Seat Leakage Past Containment Spray Heat Exchanger ESW Outlet Valve

a. Inspection Scope

On April 7, 2001, an auxiliary equipment operator noted that valve 1-WMO-717, the west containment spray heat exchanger ESW outlet valve, appeared to be leaking by. This valve was shut during normal operation to prevent excessive cooling of containment spray during the injection phase of emergency core cooling actuation. Excessive cooling of the containment spray could result in a containment pressure lower than that assumed in the 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems [ECCS] for Light-Water Nuclear Power Reactors," analysis. For the purposes of ECCS performance, lower containment pressures result in less back pressure to reactor coolant system leakage and therefore represents a greater core cooling challenge. Because this issue potentially impacts the capability of the ECCS, the inspectors determined that this condition affected the mitigating systems cornerstone. The inspectors reviewed the operability determination, the UFSAR accident analysis, and applicable procedures. The inspectors also discussed the impact of the 1-WMO-717 leakby with operations and engineering personnel.

b. Findings

No findings of significance were identified.

.2 Excessive Seat Leakage from 2-CS-300W

a. Inspection Scope

In June 2000, the licensee identified that seat leakage through 2-CS-300W, the "B" train centrifugal charging pump discharge to RCP seal water injection filter shutoff valve, resulted in a mismatch between charging and letdown flow. Leakby through the normally locked shut 2-CS-300W valve resulted in a portion of the reactor coolant seal injection flow bypassing the flow instrumentation in the normal seal injection flow path. The licensee's evaluation of this condition determined that the leakage did not result in any equipment operability impact and that repair of the condition could be deferred. Since the June 2000 Unit 2 restart, the seat leak rate through 2-CS-300W has remained constant at a rate of approximately 5 gpm. In addition to the condition evaluation for the 2-CS-300W seat leakby, the inspectors reviewed the UFSAR, Technical Specifications, and related procedures to identify potential adverse impacts from this condition. The inspectors discussed the condition with operations and engineering personnel. The inspectors determined that this operability determination was associated with the operability of the emergency core cooling system and therefore impacted the mitigating systems cornerstone.

b. Findings

No findings of significance were identified.

.3 Containment Hydrogen Sample Above Safety Analysis Limits

a. Inspection Scope

On April 26, 2001, the licensee identified that hydrogen sample results from the Unit 1 and Unit 2 containments were inconsistent with safety analysis assumptions. During preparation for planned maintenance in containment, the licensee measured maximum containment hydrogen levels of approximately 1.4 volume percent hydrogen. The UFSAR Section 14.3.6 containment hydrogen analysis assumed that initially no hydrogen was present in containment. An initial hydrogen concentration higher than assumed in the UFSAR could result in exceeding safety analysis limits. The licensee wrote condition report (CR) 01116075 and evaluated the potential operability concern. The inspectors reviewed the licensee's evaluation and the applicable UFSAR safety analysis and discussed the results of the operability determination with engineering and operations personnel. The licensee conducted followup containment hydrogen sampling with more accurate equipment and determined that the containment hydrogen level were negligible. Because excessive hydrogen in containment could challenge the containment building integrity, the inspectors determined that this issue impacted the barrier integrity cornerstone.

b. Findings

No findings of significance were identified.

.4 Operability of Unit 2 South Safety Injection Pump with Weld Leak

a. Inspection Scope

On April 12, 2001, the licensee determined that dry boric acid buildup on the downstream side of instrument isolation valve 2-IFI-266-V2 (Safety Injection line flow indicator) was due to a through wall leak in the weld where the valve is welded to the pipe. The isolation valve was closed until the weld was repaired on April 20, 2001. The inspectors reviewed the licensee’s operability evaluation which determined that the Unit 2 “B” Train mitigating system safety injection pump remained operable while the instrument was isolated awaiting repairs.

b. Findings

No findings of significance were identified.

.5 Unit 2 Station Battery

a. Inspection Scope

On April 5, 2001, during a routine surveillance, the licensee identified an unusual sediment or growth in Cell 2 of the Unit 2 “B” Train battery. The condition was documented in Condition Report 01095026. The inspectors reviewed the licensee’s operability evaluation on the identified condition of the Unit 2 “B” Train battery. The station battery is important due to its role in mitigating a station blackout event.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post maintenance testing for the following mitigating system work activities.

- On April 25, 2001, the licensee removed the Unit 2 West (“B” Train) Essential Service Water (ESW) pump from service for routine corrective and preventive maintenance work activities.
- On May 1, 2001, the licensee removed the Unit 2 AB “B” Train diesel generator from service for routine corrective and preventive maintenance work activities.
- On May 10, 2001, the licensee removed the Unit 1 TDAFWP pump from service for routine corrective and preventive maintenance work activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing of the Unit 1 West Residual Heat Removal Pump

a. Inspection Scope

On April 5, 2001, the licensee determined that the Unit 1 Train "B" residual heat removal pump exceeded its high action limit during inservice testing conducted in accordance with TS 4.0.5. The licensee reperformed the test with higher accuracy test equipment and determined that the RHR pump performed within the normal range. The inspectors reviewed the surveillance test procedure, the test data, and discussed the results of the test with operations and engineering personnel. The inspectors also reviewed the basis for RHR pump surveillance testing acceptance criteria.

b. Findings

No findings of significance were identified.

.2 Periodic Seal Injection Line Resistance Measurement

a. Inspection Scope

The inspectors reviewed the licensee implementation of TS surveillance requirement 4.6.2.1.c associated with periodic measurement of the reactor coolant pump (RCP) seal line resistance. The value of the RCP seal line resistance affected the centrifugal charging pump flow split between RCP seal injection and RCS loop cold leg injection following an engineered safety features actuation. The inspectors considered that this surveillance requirement was associated with the mitigating systems cornerstone. The inspectors reviewed surveillance procedures 01(02)-OHP 4030.STP.052L, data from recent Unit 2 seal resistance measurements, and the basis for the surveillance procedure acceptance criteria.

b. Findings

The inspectors identified two non-conservative factors used in the development of the reactor coolant pump seal line resistance measurement acceptance criteria. The first issue involved the adequacy of the design change review performed for the recent Unit 1 steam generator (SG) replacement project and the second involved the application of instrument uncertainty to the surveillance test acceptance criteria. Specifically, the inspectors identified the following issues:

- The licensee failed to identify that SG replacement non-conservatively impacted the TS 4.4.6.2.1.c seal line resistance formula. The seal injection resistance was calculated by dividing the differential pressure between the CVCS system charging header and the RCP seal injection point by the square of the total seal flow rate. The value of the RCP seal line injection point pressure, P_{si}, was assumed to be constant, but depended on several factors, including the pressure drop across the primary side of the SG. An increase in SG pressure drop decreased the value of P_{si}. The use of a lower value of P_{si} than actually existed would result in a higher calculated seal injection differential pressure and therefore over estimate the seal injection line resistance.

  The licensee determined the value of P_{si} in 1989 to support a revision to the TS 3.4.6.2 controlled leakage surveillance requirements. Since 1989, the steam generators in both Units have been replaced, resulting in lower SG pressure drops. Although the value of P_{si} should have been increased to reflect the lower SG pressure drop, the licensee failed to identify that TS 4.4.6.2.1.c was impacted by SG replacement. Consequently, the TS 4.4.6.2.1.c formula non-conservatively over-predicted seal injection line resistance by approximately 3 percent. Following the inspectors' identification of this issue, the licensee wrote CR 0117020 and issued a change to the associated surveillance test procedures to conservatively increase the value of P_{si} by 7 psi.

- The instrument uncertainty correction applied to seal injection line resistance test acceptance criteria did not bound expected test conditions. Two parameters were measured during the seal injection line resistance surveillance testing: charging header pressure and the total seal injection flow rate. Although the licensee evaluated the impact of instrument uncertainty associated with these parameters, the inspectors determined that the evaluation did not bound normal test conditions. Because of the nature of the seal injection resistance formula, the use of either a low charging header pressure or high seal injection flow rate in the uncertainty evaluation tended to reduce the impact of instrument uncertainty.

  The licensee based their uncertainty evaluation on a nominal charging header pressure of 2419 psig and a seal injection line flowrate of 40 gpm. The inspectors determined, based on test data, that actual charging header pressure was approximately 20 psi higher than the assumed nominal value. Additionally, the surveillance procedure allowed the seal injection flow rate to be as low as 24 gpm during testing. The inspectors performed an independent uncertainty calculation and determined that the licensee's uncertainty methodology could under-predict instrument uncertainty by as much as 50 percent. The overall impact of under-predicting the impact of instrument uncertainty was that the test acceptance criteria could have been established too low to ensure TS compliance under all expected test conditions. The licensee initiated CR 01124053 to evaluate the uncertainty correction contained in the RCP seal injection line resistance measurement surveillance procedures.

10 CFR 50, Appendix B, Criterion XI, "Test Control," required, in part, a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the licensee failed to ensure that the acceptance criteria contained in test procedures 01(02)-OHP 4030.STP.052L adequately incorporated limitations associated with steam generator replacement and instrument uncertainty. Specifically, the value of the parameter P_{si} did not include consideration of the replacement steam generator lower differential pressure and the test acceptance criteria did not adequately consider instrument uncertainty over the range of expected test conditions. The inspectors concluded that this failure constituted a **Non-Cited Violation (50-315/01-09-01, 50-316/01-09-01)** of 10 CFR 50, Appendix B, Criterion XI, consistent with Section VI.A. of the NRC Enforcement Policy. This violation is in the licensee's corrective action system as CR 01117020 and CR 01124053. This NCV is closed.

The inspectors evaluated the risk significance of this issue using the Significance Determination Process. The value of the seal line resistance impacts the capability of the high head emergency core cooling system and therefore was associated with the mitigating systems cornerstone. The inspectors determined that the failure to adequately recognize the impact of significant plant modifications and instrument uncertainty on technical specification requirements could become a more significant safety concern if left uncorrected and, therefore, was more than a minor concern.

Based on a review of recent test data, the inspectors determined that the impact of this failure was bounded by existing margin. Consequently, this issue did not result in inoperability of the controlled leakage charging flow path and was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review.

.3 Type C Containment Leak Rate Testing

a. Inspection Scope

Technical Specification 3.6.1.2, "Containment Leakage," required measurement of the leakage for all penetrations and valves subject to type B & C testing in accordance with 10 CFR 50, Appendix J. Because containment leakage can result in increased post-accident dose to plant operators and members of the public, the inspector determined that this surveillance requirement was associated with the barrier integrity cornerstone. The inspectors reviewed the type B & C testing procedure, discussed the Appendix J testing program with engineering personnel, and reviewed the results from recently completed testing for Unit 1.

b. Findings

No findings of significance were identified.

.4 Surveillance Test of Unit 1 Turbine Driven Auxiliary Feedwater Pump

a. Inspection Scope

The inspectors reviewed surveillance test 01-OHP.4030.STP.017T performed May 10, 2001. The inspectors reviewed the test data and verified that the selected surveillance test met the TS and licensee procedural requirements. The inspectors discussed these surveillance tests with operations, engineering, and regulatory affairs personnel.

b. Findings

No findings of significance were identified.

.5 Functional Test of Unit 1 Power Range Nuclear Instrument N-42

a. Inspection Scope

On May 9, 2001, the inspectors observed a functional check of Unit 1 power range nuclear instrument N-42. The functional check was done to satisfy the monthly surveillance requirement of Technical Specification 4.3.1.1.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed the licensee's gathering and submittal of data for the following Unit 2, first quarter of 2001, information:

- Unplanned Scrams per 7,000 Critical Hours portion of the Initiating Events cornerstone
- Scrams with Loss of Normal Heat Sink

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 (Discussed) FIN 50-315/01-07-02: Failure to implement adequate corrective actions for previously identified Maintenance Rule violations

a. Inspection Scope

The inspectors assessed the effectiveness of the licensee's corrective actions for previous Maintenance Rule violations during the reviews of Maintenance Rule implementation documented in Section 1R12, above. In NRC Inspection Reports 50-315/00-20; 50-316/00-20 and 50-315/00-22; 50-316/00-22, issued in October and December 2000, respectively, the inspectors documented previous violations of Maintenance Rule requirements. These violations were associated with the licensee's failure to properly identify and evaluate Maintenance Rule functional failures (NCV 50-315/00-20-03; 50-316/00-20-03), and the failure to monitor the Maintenance Rule unavailability of systems required during shutdown mode operation (NCV 50-315/00-20-01; 50-316/00-20-01). In response to these violations, the licensee implemented a corrective action plan which included training for personnel responsible for implementing the Maintenance Rule program and historical reviews of reliability failures and SSC unavailability.

b. Findings

In NRC Inspection Report 50-315/01-07; 50-316/01-07, the inspectors identified finding (FIN) 50-315/01-07-02 associated with the licensee's failure to implement adequate corrective actions for previously identified Maintenance Rule violations. During the Maintenance Rule implementation inspection efforts documented in Section 1R12 above, the inspectors identified additional examples of weak corrective actions. These examples involved the failure to adequately trend system performance against performance criteria, evaluation of potential Maintenance Rule functional failures, and the tracking of system unavailability. Specifically, the inspectors identified the following issues:

- A total of eight CIV Appendix J leak rate testing failures were not adequately reviewed against the expert panel approved performance criteria. The performance criteria for the CIV system required that, if greater than 5 percent of the containment isolation valves failed to meet Appendix J testing leakage limits, the CIV system would be presented to the expert panel for (a)(1) consideration. The licensee failed to identify that these eight test failures potentially impacted the CIV performance criteria. After the inspectors questioned the Maintenance Rule monitoring and trending of Appendix J test failures, the licensee determined that the CIV system performance criteria was potentially exceeded. The licensee initiated CR 01122038 to evaluate this condition.

- The licensee failed to properly account for approximately ten hours of emergency boration unavailability time which occurred on February 21, 2001. The emergency boration function was monitored with both reliability and unavailability performance criteria. Although the unavailability time was documented in the control room log, engineering personnel failed to identify this system unavailability time. The licensee determined that this additional unavailability time would not have caused the function to exceed its performance criteria. The licensee initiated CR 01123103 to further evaluate this condition.

- A total of three potential CVCS functional failures associated with makeup and inventory control functions were not adequately evaluated for Maintenance Rule impact. The licensee either failed to identify that a Maintenance Rule function was affected by the failure or applied inappropriate credit for operator action in maintaining functional capability. The licensee determined that the addition of these potential functional failures to previously identified reliability failures would not have resulted in exceeding the associated performance criteria. The licensee appropriately identified these issues in the corrective action system for further evaluation.

The inspectors determined that the circumstances and causes of the above failures were similar to the Maintenance Rule corrective action weaknesses identified in FIN 50-315/01-07-02. These issues were significant in that the failure to identify adverse reliability and unavailability trends could result in degrading system performance. However, the inspectors concluded that the current Maintenance Rule implementation weaknesses were additional examples of previously identified implementation weaknesses and did not represent new licensee performance issues. In order to improve the quality of Maintenance Rule evaluations, the licensee formed the Plant Engineering Review Committee to review and approve Maintenance Rule failure evaluations. The licensee addressed additional Maintenance Rule corrective actions in Engineering Action Plan 01-614, including additional training for personnel performing Maintenance Rule failure evaluations. The inspectors determined that the licensee's planned actions to address Maintenance Rule implementation weaknesses were reasonable.

4OA3 Event Follow-Up (71111.14 and 92700)

.1 Licensee Event Reports

a. Inspection Scope

The inspectors reviewed the corrective actions associated with the following licensee event reports.

b. Findings

(Closed) Licensee Event Report 50-315/97011-03: Operation outside design basis for ECCS [emergency core cooling systems] and containment spray pumps for switchover to recirculation sump suction. This issue was previously identified as Unit 2 Restart Action Matrix (RAM) Item R2.3.2 which was closed in NRC Inspection Report 50-315/00-13; 50-316/00-13. Revision 3 to the LER documented the licensee's completed root cause evaluation. However, the LER supplement did not identify any new issues; therefore, this LER is closed.

(Closed) Licensee Event Report 50-315/97025-00: Unexpected ESF [Engineered Safety Features] actuation during filling SG for wet layup. On September 23, 1997, with Unit 1 in Mode 5, an unplanned ESF actuation occurred while filling the steam generators to wet layup. This event was discussed in NRC Inspection Report 50-315, 50-316/1997-015, Section O2.2 and documented in CR 97-2596. The licensee determined that the cause of the ESF actuation was oscillations in steam generator level resulting in activation of the low low steam generator water level auxiliary feedwater system actuation. The inspectors reviewed the LER, corrective actions, and associated procedures and identified no additional issues. This LER is closed.

(Closed) Licensee Event Report 50-315/97028-00: Failure to comply with 10 CFR 50, Appendix R requirements results in unanalyzed condition. During the Thermo-Lag resolution, a concern was identified that the fire stops installed on the cable trays traversing from the north side (Fire Area 44N) to the south side (Fire Area 44S) in Fire Zone 44 did not meet the requirements of 10 CFR 50, Appendix R, Section III.G.2.b, which required separation of cables of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles. It was mistakenly assumed that fire stops were used to establish the 20 feet separation so the fire area would meet Section III.G.2 requirements. Further review by the licensee indicated that the fire area was not initially proposed and designed as a fire area meeting Section III.G.2 requirements.

In the licensee's March 31, 1983, letter to the NRC, the licensee had described the safe shutdown assessment for the plant. Fire Areas 44N and 44S were identified as meeting the requirements of 10 CFR 50, Appendix R, Section III.G.3. The letter further stated that certain cable trays would have one-hour rated barriers installed in Fire Zone 44. The open cable trays traversing Fire Zone 44 from the north side to the south side would be appropriately fire stopped to prevent fire propagation from one section of the fire zone to the other. The initial purpose of the fire stop was to establish separation of fire zones so that a fire would not propagate to the other unit whose equipment was relied upon for safe shutdown of the fire-affected unit. On November 22, 1983, the NRC issued a safety evaluation report which stated that the licensee's proposed alternate shutdown capability (with references to the March 31, 1983 letter) complied with the requirements of Section III.G and III.L of Appendix R. The NRC did not take any exception to the method which the licensee used to establish area independence between Fire Areas 44N and 44S. Therefore, the installation of the fire stops on the traversing cables trays from north to south in Fire Zone 44 met the requirements of Appendix R, Section III.G.3. This LER is closed.

<u>(Closed) Licensee Event Report 50-315/98001-03</u>: Containment air recirculation system flow testing results indicate condition outside the design basis. The issue described in this licensee event report was the subject of Unit 2 RAM Item R2.1.2 and EEI 50-315/98007-06; EEI 50-316/98007-06, which were closed in NRC Inspection Report 50-315/99029; 50-316/99029. The licensee wrote CR 98-1017 to document the issue. Revision 3 to the LER documented the licensee's completed root cause evaluation. However, the LER supplement did not identify any new issues; therefore, this LER is closed.

<u>(Closed) Licensee Event Report 50-315/98018-03</u>: Retraction - Use of reactor coolant pump seals as alternate boron injection path. The inspectors reviewed the licensee's design basis documentation and determined that the use of reactor coolant pump seals as an alternate boron injection path was not part of the design basis of the plant. Because a condition outside the design basis of the plant did not exist, the condition was not reportable; therefore, this LER is closed.

<u>(Closed) Licensee Event Report 50-315/98023-00,-01,-02</u>: Retraction - Potential single failure due to cross train routing of non-safety related cables. During the inspection of Unit 2 Case Specific Checklist Item 7, "Resolution of Non-Safety Related Cables Going to Shunt Trip Coils," the NRC determined that the plant design met the licensing basis for load shed circuitry and balance of plant loads. This determination was documented in NRC Inspection Report 50-315/00-13; 50-316/00-13. Because a condition outside the design basis of the plant did not exist, the condition was not reportable; therefore, this LER is closed.

<u>(Closed) Licensee Event Report 50-315/98028-00</u>: Technical Specification surveillance requirement not met due to lack of understanding of "Staggered Test Basis." This LER reported that the licensee failed to test the containment air recirculation (CEQ) system on a staggered test basis as required by Technical Specification 4.6.5.6. After investigating this issue, the licensee determined that failure to test the CEQ system at the proper interval did not prevent the detection of an inoperable CEQ train. This condition was corrected as part of the licensee's restart effort after the licensee reviewed and revised the surveillance testing program. The inspectors review of the licensee's surveillance testing program was documented in NRC Inspection Report 50-315/99033; 50-316/99033. Unit 2 RAM Item 1, "Programmatic Breakdown in Surveillance Testing," was closed in NRC Inspection Report 50-315/00-01; 50-316/00-01. Therefore, this LER is closed.

(Closed) Licensee Event Report 50-315/98047-01: Reactor coolant pump nitride seals. This issue was previously identified as Unit 2 RAM Item R.2.3.55 which was closed in NRC Inspection Report 50-315/00-16; 50-316/00-16. Revision 1 to the LER documented the licensee's root cause evaluation. However, the LER supplement did not identify any new issues; therefore, this LER is closed.

(Closed) Licensee Event Report 50-315/98049-01: Emergency boration flowpath inoperable due to original design deficiency. In October 1998, the licensee identified that the design of the emergency boration system was inadequate. Supplement 1 to this LER provided additional information relating to the description, cause, and corrective actions for this design deficiency. Technical Specifications 3.1.2.5 and 3.1.2.6, "Boric Acid Transfer Pumps," required an operable boric acid transfer pump to support the boron injection flowpath from the boric acid storage tanks. The licensee identified three design issues in this LER: (1) inadequate net positive suction head for the boric acid transfer pumps, (2) potential to exceed reactor coolant pump seal temperature and boric acid concentration design limitations, and (3) potential for borated water to be subject to temperatures below the solubility limit in portions of the chemical and volume control system. The licensee's corrective actions, as documented in CR 98-5914, included a design change package to reduce the boric acid concentration from a nominal value of twelve weight percent to four weight percent in addition to programmatic improvements in the design control program.

10 CFR 50, Appendix B, Criterion III, "Design Control," required, in part, that measures shall be established to assure that the design basis for systems, structures, and components, are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee failed to ensure that the emergency boration flow path could be operated within its design basis. Specifically, the design of the emergency boration flowpath could potentially result in: (1) inadequate net positive suction head for the boric acid transfer pumps, (2) a potential for exceeding design limitations for the RCP seals, or (3) the failure to maintain system temperatures above applicable solubility limits. The inspectors determined that this issue could have had a credible impact on safety and therefore was more than a minor concern. The inspectors concluded that this failure constituted a **Non-Cited Violation (50-315/01-09-02; 316/01-09-02)** of 10 CFR 50, Appendix B, Criterion III consistent with the NRC Enforcement Policy. This violation is in the licensee's corrective action system as CR 99-5914 and CR 98-0876. The NRC Inspection Manual Chapter 0350 panel and the Region III Senior Reactor Analysts reviewed and assessed the risk significance of this item as Unit 2 Restart Action Matrix Item R.2.3.56. This item was determined to be a low priority issue of very low risk significance. This LER and NCV are closed.

(Closed) Licensee Event Report 50-315/98051-00,-01: Reactor trip signal from manual safety injection not verified as required by Technical Specification surveillance. On November 22, 1998, the licensee identified that Unit 1 reactor trip breakers had not been tested in accordance with TS 4.3.2.1.1. This Technical Specification, which was applicable in Modes 1 through 4, required, in part, that the reactor trip signal from manual safety injection be verified. The inspectors concluded that the LER documented a violation of TS 4.3.2.1.1. However, the failure to verify the reactor trip signal from a manual safety injection prior to entering Mode 4 (Hot Shutdown) was of minimal safety significance in that the control rod drive system was not made capable of rod withdrawal until after the unit was placed in Mode 3 (Hot Standby). This issue was identified as Unit 2 RAM Item R.2.1.18, which was closed in NRC Inspection Report 50-315/00-01; 50-316/00-01. The licensee entered the failure to perform reactor trip breaker testing in Condition Report 98-6496. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

(Closed) Licensee Event Report 50-315/98059-01: Single failure in containment spray system could result in pH outside design. The issue described in this licensee event report was the subject of Unit 2 RAM Item R2.3.60 which was closed in NRC Inspection Report 50-315/99029; 50-316/99029. The licensee included this issue and other containment sump pH issues in the corrective action program as CR 98-7575 and CR 99-6468. Revision 1 to the LER did not identify any new issues; therefore, this LER is closed.

4OA5 Other

(Closed) URI 50-315/99007-05(DRS); 50-316/99007-05(DRS): 600 Volt Alternating Current (VAC) Cable Sizing. The licensee had not documented which of the three National Electrical Manufacturers Association (NEMA) cable insulation levels applied to the 600 VAC system in an ungrounded configuration. The licensee reviewed the 600 VAC ungrounded system design and determined that the utilized cables were rated at 600 VAC and had been procured to conform to the specifications issued by NEMA. The licensee believed that the three insulation levels were proposed, but not finalized as a NEMA standard at the time the cables were procured. The licensee performed an engineering study and concluded the procured cables for the 600 VAC system had an insulation thickness that was equivalent to the 173 percent level specified by NEMA. The licensee indicated this was the appropriate insulation level based on their protective scheme for a cable section becoming grounded.

This issue was entered into the licensee's corrective action program under CR 99-19650. The inspectors reviewed the condition report and concluded that the corrective actions were appropriate for closure of the issue. This item is closed.

4OA6 Management Meetings

The inspectors presented the inspection results to licensee management listed below on May 16, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

R. Crane, Regulatory Affairs, Compliance Supervisor
R. Gaston, Regulatory Affairs, Manager
S. Greenlee, Director, Design Engineering and Regulatory Affairs
M. Hoskins, System Engineering
J. Johns, Maintenance Rule Program Owner
S. Lacey, Director, Plant Engineering
J. LaPlante, Performance Assurance Manager
J. Mathis, Regulatory Affairs
R. Meister, Regulatory Affairs
J. Molden, Maintenance Department Director
D. Moul, Assistant Operations Superintendent
T. Noonan, Director, Performance Assurance
S. Partin, Assistant Operations Manager
J. Piazza, Chemistry Supervisor
J. Pollock, Plant Manager
R. Powers, Senior Vice President
M. Rencheck, Vice President, Nuclear Engineering
L. Weber, Manager, Operations

NRC

A. Vegel, Chief, Reactor Projects Branch 6
J. Stang, Project Manager, NRR
G. Grant, Director, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/01-09-01 50-316/01-09-01	NCV	Non-conservative acceptance criteria used in seal injection line resistance surveillance procedure (Section 1R22)
50-315/01-09-02 50-316/01-09-02	NCV	Emergency boron injection path inoperable due to original design deficiencies (Section 4OA3)

Closed

50-315/97011-03	LER	Operation outside design basis for ECCS and containment spray pumps for switchover to recirculation pump suction (Section 4OA3)
50-315/97025-00	LER	Unexpected ESF actuation during filling steam generators for wet layup (Section 4OA3)

50-315/97028-00	LER	Failure to comply with 10 CFR 50 Appendix R requirements results in unanalyzed condition (Section 4OA3)
50-315/98001-03	LER	Containment air recirculation system flow testing results indicate condition outside the design basis (Section 4OA3)
50-315/98018-03	LER	Retraction - Use of reactor coolant pump seals as alternate injection path (Section 4OA3)
50-315/98023-02	LER	Retraction - Potential single failure due to cross train routing of non-safety related cables (Section 4OA3)
50-315/98028-00	LER	Technical Specification surveillance requirement not met due to lack of understanding of “Staggered Test Basis” (Section 4OA3)
50-315/98047-01	LER	Reactor coolant pump nitride seals (Section 4OA3)
50-315/98049-01	LER	Emergency boron injection path inoperable due to original design deficiencies (Section 4OA3)
50-315/98051-01	LER	Reactor trip signal from manual safety injection not verified as required by Technical Specification surveillance (Section 4OA3)
50-315/98059-01	LER	Single failure in containment spray system could result in pH outside design (Section 4OA3)
50-315/99007-05 50-316/99007-05	URI	600 Volt Alternating Current Cable Sizing (Section 4OA5)
50-315/01-09-01 50-316/01-09-01	NCV	Non-conservative acceptance criteria used in seal injection line resistance surveillance procedure (Section 1R22)
50-315/01-09-02 50-316/01-09-02	NCV	Emergency boron injection path inoperable due to original design deficiencies (Section 4OA3)

Discussed

50-315/01-07-02	FIN	Failure to implement adequate corrective actions for previously identified Maintenance Rule violations

LIST OF ACRONYMS USED

AES	Engineered Safety Features Ventilation
AFW	Auxiliary Feedwater System
ATR	Administrative Technical Requirement
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CR	Condition Report
CTS	Containment Spray System
CVCS	Chemical and Volume Control System
D/G	Diesel Generator
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Features
ESW	Essential Service Water
FIN	Finding
IMC	Inspection Manual Chapter
JO	Job Order
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MDAFWP	Motor Driven Auxiliary Feedwater Pump
MHP	Maintenance Head Procedure
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODE	Operability Determination Evaluation
OHI	Operations Head Instruction
OHP	Operations Head Procedure
OSO	Operations Standing Order
OWA	Operator Workaround
PDR	Public Document Room
PI	Performance Indicator
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PMT	Post-maintenance Testing
PORV	Power Operated Relief Valve
PPC	Plant Process Computer
PRT	Pressurizer Relief Tank
RAM	Restart Action Matrix
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SG	Steam Generator
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TDAFWP	Turbine Driven Auxiliary Feedwater Pump

TS	Technical Specification
URI	Unresolved Item
UFSAR	Updated Final Safety Analysis
VAC	Volts, Alternating Current
VDC	Volts, Direct Current
VIO	Violation

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

.1 Unit 1 Component Cooling Water

UFSAR Section 9.5	Component Cooling System	
Unit 1 TS 3.7.3	Component Cooling Water System	
01-OHP 4021.016.001	Filling and Venting the Component Cooling Water System	Revision 11b
01-OHP 4021.016.003	Operation of the Component Cooling Water System During System Startup and Power Operation	Revision 15a
01-OHP 4022.016.001	Malfunction of the CCW System	Revision 2
01-OHP 4022.016.004	Loss of Component Cooling Water	Revision 5
Flow Diagram OP-1-5135 (series)	CCW Pumps and CCW Heat Exchangers	
CR 01122070	NRC identified procedure and labeling differences in CCW lineup procedure	May 2, 2001
CR 01129004	North spent fuel pit heat exchanger outlet valve, 1-CCW-114, has inconsistent configuration control guidance	May 9, 2001
CR 01129052	NRC identified that ESW drain valves on CTS heat exchangers have caps installed instead of hose connections as shown on the flow diagram	May 9, 2001
CR 01135029	NRC identified valves locations for CCW system listed wrong on lineup sheet	May 15, 2001

.2 Partial Equipment Alignment of Unit 1 East Motor Driven Auxiliary Feedwater Pump

UFSAR Section 10.5.2	Auxiliary Feedwater System	
01-OHP 4021.056.001	Filling and Venting Auxiliary Feedwater System	Revision 20
01-OHP 4021.056.002	Auxiliary Feed Pump Operation	Revision 18
NUREG/CR-5832	Auxiliary Feedwater System Risk-Based Inspection Guide for the D. C. Cook Nuclear Power Plant	

PMP 4043.SLV.001	Seal/Locked Valves	Revision 4
PMP 4043.VLU.001	Valve Lineup and Position Control	Revision 1
Flow Diagram OP-1-5106A	Auxiliary Feedwater	Revision 49
CR 98-05865	High Point Vent Located in the Pump Room is not used to Vent System	October 15, 1998
CR 99-14763	Several answers to the 1998 NRC AFW Safety System Functional Inspection are Inadequate	June 7, 1999

.3 Partial Equipment Alignment of Unit 2 Control Air

UFSAR Section 9.8.2	Compressed Air System	
02-OHP 4021.064.001	Operation of Plant and Control Air	Revision 11
02-OHP 4022.064.001	Control Air Malfunction	Revision 4
CR 01025067	Unit 2 East control air dryer intermittently brings in annunciator when it shifts to the right tower in service	January 25, 2001
CR 01086067	Procedures have conflicting positions for Non-Essential Service Water Valves	March 27, 2001
CR 01097005	Gross air leak on north pipe union for 2-XSO-56	April 7, 2001
CR 01129093	Procedure 02-OHP 5030.064.001, control air compressor functional inspection, needs procedural improvement	May 9, 2001

.4 Unit 1 Control Room Air Conditioning

01-OHP 4021.028.014	Operation of the Control Room Air Conditioning and Pressurization/Cleanup Filter Systems	Revision 13
Flow Diagram OP-1-5149	Control Room Ventilation, Unit 1	
	Unit 1 Control Room logs	December 11, 2000 through April 19, 2001
CR 01034002	Valve 1-DW-157S leaks by its seat	February 3, 2001

1R05 Fire Protection

.1 Routine Fire Zone Tours

UFSAR Section 9.8.1	Fire Protection System	
	D. C. Cook Nuclear Plant Fire Hazards Analysis, Units 1 and 2	Revision 8
	D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook	February 1995
ATR 1-FP-1	Unit 1 Fire Detection	
PMP 2270.CCM.001	Control of Combustible Materials	Revision 0
PMP 2270.FIRE.002	Responsibilities for Cook Plant Fire Protection Program Document Updates	Revision 0
PMP 2270.WBG.001	Welding, Burning and Grinding Activities	Revision 0
PMI 2270	Fire Protection	Revision 26

.2 Annual Fire Drill Observation

CR 01109020	Fire drill conducted in west diesel fire pump room revealed that excess fire hose is provided in fire pump house hallway and that evaluation of foam provisions is desirable	April 18, 2001

1R11 Licensed Operator Requalification

02-OHP 4025.001.001	Emergency Remote Shutdown	Revision 3
02-OHP 4023.E-0	Reactor Trip or Safety Injection	Revision 16b

1R12 Maintenance Rule Implementation

.1 Containment Isolation Valve System

	Donald C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment Final Report, Containment Isolation Analysis Notebook	Revision 0 April 1992

	Containment Isolation Valve Super System Maintenance Rule Scoping	November 2000
CR 01082030	Documentation of historical review of job orders and logs for impact on containment isolation valve Maintenance Rule functions	March 23, 2001
CR 00-10442	1-XCR-100 exceeded administrative permissible leakage limit during Appendix J testing	July 25, 2000
CR 00-10831	1-NS-283 exceeded administrative permissible leakage limit during Appendix J testing	August 2, 2000
CR 00-10943	1-ECR-32 exceeded administrative permissible leakage limit during Appendix J testing	August 5, 2000
CR 00-11195	1-N-102 exceeded administrative permissible leakage limit during Appendix J testing	August 11, 2000
CR 00-11326	1-DCR-610 exceeded administrative permissible leakage limit during Appendix J testing	August 14, 2000
CR 00-11327	1-DCR-611 exceeded administrative permissible leakage limit during Appendix J testing	August 14, 2000
CR 00-11541	1-ICM-265 exceeded administrative permissible leakage limit during Appendix J testing	August 18, 2000
CR 00-11647	1-SI-189 exceeded administrative permissible leakage limit during Appendix J testing	August 22, 2000
CR 00287054	1-N-160 exceeded the guideline leakage limit during Appendix J testing	October 13, 2000
CR 00336085	1-WCR-947 has a body to bonnet leak of about 10 ounces per minute when closed	December 1, 2000
CR 00301039	1-CCW-244-72 exceeded administrative permissible leakage limit during Appendix J testing	October 27, 2000

CR 01122039	NRC identified potential non-conservatism in containment isolation valve performance criteria	May 2, 2001
CR 01122038	NRC identified that the Maintenance Rule condition monitoring for the Units 1 and 2 containment isolation valve system was not adequately trended	May 2, 2001

.2 Chemical and Volume Control System

	Chemical and Volume Control System Maintenance Rule Scoping Document	April , 2001
CR 003544057	Documentation of historical review of job orders and logs for impact on charging letdown and emergency boration system	December 19, 2000
	Maintenance Rule (a)(1) Action Plan for D.C. Cook Unit 1 and Unit 2 Chemical and Volume Control System	February 2001
	System Health Report, Charging, Letdown, Emergency Boration, Cook Unit 2	September 1, 2000 through December 31, 2000
	Control Room Logs, Unit 2, February 2001	
Flow Diagram OP 12-5131	CVCS-Boron Makeup, Units 1 & 2	
Flow Diagram OP 2-5129A	CVCS-Reactor Letdown and Charging Unit No. 2	
Flow Diagram OP 2-5129	CVCS-Reactor Letdown and Charging Unit No. 2	
Engineering Action Plan 00-395	CVCS Cross Tie Valves	
CR 00339006	1-QRV-412 did not open automatically during the initiation of a manual blend	December 4, 2000
CR 00350042	1-QRV-303 popped open/leaked by when placed in automatic following performance of a reactor coolant system leak rate surveillance test	December 15, 2000

CR 00291057	With primary water demand set at 0 gallons, 2-QRV-422 opened and approximately 14 gallons of primary water were added to the VCT during a manual boration of the VCT	October 17, 2000
CR 00-6822	Approximately 15 gpm leakage from Unit 1 to Unit 2 was identified during CVCS cross connect valve testing	May 11, 2000
CR 01123103	NRC identified that the CVCS system manger failed to identify maintenance rule unavailability for the emergency boration function that had occurred on February 21, 2001	May 3, 2001
CR 01123100	NRC identified that the documented monitoring goal for the emergency boration function was inconsistent with actual plant practice	May 3, 2001
CR 01123106	NRC identified that Maintenance Rule evaluation for CR 00350042, associated with 1-QRV-303 leakby, improperly credited operator action	May 3, 2001
CR 01122063	NRC identified that Maintenance Rule evaluation was not performed for CR 00339006, which documented a failure of 1-QRV-412, title, to open during CVCS blender operations	May 2, 2001
CR 01122065	NRC identified that Maintenance Rule evaluation was not performed for CR00291057, which documented an undemanded addition of primary water to the VCT	May 2, 2001

.3 Component Cooling Water Train Isolation Valves (Both Units)

	Component Cooling Water Maintenance Rule Scoping Document
UFSAR Section 9.5	Component Cooling Water System
Unit 1 and Unit 2 TS 3.7.3.1	Component Cooling Water System

	Control Room logs	December 1, 2000 through May 1, 2001
WR A204980	1-CMO-416 did not close with the control switch in the closed position	
JO C57154	Refurbish 1-CMO-416 actuator	September 7, 2000
CR 98-7707	While attempting to drain CCW surge tank, the indicated level stopped decreasing	December 7, 1998
CR 99-2940	Post maintenance testing requirements did not test for deficiency	February 17, 1999
CR 00-9549	While hanging clearance 1002175, CMO-416 control switch was placed in close but the valve stayed open	July 4, 2000
CR 00311027	Several valves in both AFW and CCW with closed safety functions were identified that may have not been evaluated for seat leakage versus gross leakage capability	November 2, 2000
CR 00356032	Component Cooling Water System Maintenance History Review	December 21, 2000

1R13 Maintenance and Emergent Work (71111.13)

.1 Unit 2 West Essential Service Water Pump Outage

	Work Week Cycle 36, W-11, On-Line Work Schedule Review Risk Assessment	April 11, 2001
PMP-2291.OLR.001	On-Line Risk Management	Revision 1
CR 01115053	Procedural Inadequacies in Essential Service Water System Operating Procedure associated with removal of an ESW loop from service	April 25, 2001
ATR 2-ESW-1	Essential Service Water System	
PA-00-01EVAL	D.C. Cook 12-Week Maintenance Schedule Risk Evaluation	Revision 0

.2 Maintenance During the Week of April 15, 2001

PMP 2291.OLR.001	On-Line Risk Management	Revision 0
	PMP-2291.OLR.001, On-Line Risk Management Work Schedule Review and Approval For Cycle 36, Week 10 Work Week Schedule	
	PRA Analysis Summary for Cycle 36, Week 10 Work Week Schedule	
	Control Room logs	April 15 through April 21, 2001

.3 Maintenance During the Week of April 29, 2001

PMP 2291.OLR.001	On-Line Risk Management	Revision 0
	PMP-2291.OLR.001 On-Line Risk Management Work Schedule Review and Approval For Cycle 36, Week 12 Work Week Schedule	
	PRA Analysis Summary for Cycle 36, Week 12 Work Week Schedule	
	Control Room logs	April 29 through May 5, 2001

.4 Unit 1 Turbine Driven Auxiliary Feedwater Pump

Unit 1 TS 3.7.1.2	Auxiliary Feedwater System	
CL #1004959	Clearance on the Unit 1 TDAFP	May 9, 2001
JO R212011	Unit 1 TDAFP, lubricate pump bearings and coupling, sample oil	May 10, 2001
01-OHP 4030.STP.017T	Turbine Driven Auxiliary Feedwater System Test	Revision 15
	Unit 1 Control Room logs	May 9 and May 10, 2001
CR 01130056	Unit 1 TDAFP mechanical overspeed trip lever did not trip the trip and throttle valve as expected during the performance of 01-OHP 4030.STP.017T	May 10, 2001

.5 Unit 2 “B” Train Emergency Diesel Generator Maintenance Outage

UFSAR Section 8.4	Emergency Power System	
Unit 2 TS 3.8.1	AC Sources - Operating	
PMP 2291.OLR.001	On-Line Risk Management	Revision 1
JO C206083	Correct jacket water leakage on Unit 2 AB emergency diesel generator at #1 front bank cylinder	May 3, 2001
CR 00-9516	Unit 2 AB emergency diesel generator jacket water leak on liner to front bank cylinder head #1	July 3, 2000

1R15 Operability Evaluations

.1 Excessive Seat Leakage Past Containment Spray Heat Exchanger ESW Outlet Valve

UFSAR Section 14.3.1	Large Break Loss-of-Coolant-Accident Analysis	
NUREG-0800, Section 6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	Revision 2 July 1981
Branch Technical Position CSB-1	Minimum Containment Pressure Model for PWR ECCS performance	Revision 2 July 1981
01-OHP 4024.138	Annunciator #138 Response: RMS [Radiation Monitoring System] Electro-Larm,"	Revision 7
DIT B-2016	Reduced Spray Temperature Impact on LOCA PCT [Peak Centerline Temperature] for ODE [Operability Determination Evaluation]	Revision 1
CR 01097018	1-WMO-717, the west containment spray heat exchanger ESW outlet valve, appears to be leaking by	April 7, 2001

.2 Excessive Seat Leakage from 2-CS-300W, title

Evaluation MD-02-CVCS-049-N	Determine Impact of 2-CS-300W Leaking on U2 Charging Flow Balance

Flow Diagram OP-2-5129	CVCS-Reactor Letdown and Charging Unit No. 2	
CR 00-8054	Mismatch of 20 gpm between charging and letdown flow	June 3, 2000
CR 00-8103	2-CS-300W, west centrifugal charging pump discharge to RCP seal water injection filter shutoff valve, leaks by	June 4, 2000

.3 <u>Containment Hydrogen Sample Above Safety Analysis Limits</u>

UFSAR Section 14.3.6	Hydrogen in Containment After a Loss-of-Coolant Accident	
Regulatory Guide 1.7	Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident	Revision 2, November 1978
DIT B-2034	Containment Hydrogen Concentration	Revision 0
CR 01116075	Hydrogen sampling results for the Unit 1 containment was approximately 1.37% and the sampling results for Unit 2 were 0.7 to 0.8% hydrogen	April 26, 2001

.4 <u>Operability of Unit 2 South Safety Injection Pump with Weld Leak</u>

UFSAR Table 7.8-1	Type “A” Variables Provided the Operator for Manual Functions During and Following an Accident	
UFSAR Table 14.4.2-1	Equipment Required to Shutdown Reactor (Unit 2) (for High Energy Line Break Ruptures Outside the Containment)	
Unit 2 TS 3.4.10.1	Structural Integrity	
	Emergency Operating Procedures	
FO-01-D-051	Performance Assurance Field Observation, 2-IFI-266 Weld Leak Operability Review	
CR 01103002	Dry boric acid buildup on downstream side of valve where tubing is welded to valve 2-IFI-266-V2	April 13, 2001

.5 Unit 2 Station Battery

Unit 2 TS 3/4.8.2.3	D.C. Distribution - Operating	
UFSAR Section 8.3.4	250 Volt DC System	
12-IHP 4030.STP.601	AB, CD and N-Train Battery Quarterly Surveillance and Maintenance	Revision 3
JO R58875	Perform 2-BATT-N 60 month surveillance	September 16, 1999
JO R213103	Perform 2-BATT-AB 92 day surveillance	April 5, 2001
CR 01087069	While performing a modified performance test for 2-BATT-N, test result data was incorrectly entered on the associated data sheet, potentially calling into question the operability of 2-BATT-N	March 28, 2001
CR 01095026	Unit 2 battery 2-BATT-AB has an abnormal growth on one of the positive plates of cell #2	April 5, 2001

1R19 Post Maintenance Testing

.1 Post Maintenance Testing Following Corrective Maintenance on the Unit 2 West Essential Service Water Pump

Unit 2 TS 3.7.4.1	Essential Service Water System	
PMP 2291.PMT.001	Work Management Post Maintenance Testing Matrices	Revision 2
JO R0098845	Verify proper operation of vent trap	May 3, 2001
JO R0213634	Perform PMT for sensing lines to discharge strainer differential pressure switch	April 25, 2001

.2 Post Maintenance Testing Following Maintenance on the Unit 1 TDAFWP

UFSAR Section 10.5.2	Auxiliary Feedwater System	
Unit 2 TS 3.7.1.2	Auxiliary Feedwater System	
01-OHP.4030.STP.017T	Turbine Driven Auxiliary Feedwater System Test	Revision 15
JO R0212011	PMT following lubrication of TDAFWP coupling	May 10, 2001

.3 Unit 2 “B” Train Emergency Diesel Generator Maintenance Outage

PMP 2291.PMT.001	Work Management Post Maintenance Testing Matrices	Revision 2
02-OHP 4030.STP.027AB	AB Diesel Generator Operability Test (Train B)	Revision 16
JO C192411	Replace broken four point terminal block	May 13, 2001
JO R214995	Slow speed start of Unit 2 AB emergency diesel generator	May 15, 2001
CR 99-25782	Broken terminal block in breaker cubicle 2-ABD-A-2A, Unit 2 AB emergency diesel bypass lube oil filter pump	October 21, 1999
CR 01122034	Unit 2 AB emergency diesel generator #6 rear bank fuel injector pump is leaking at the pipe fitting	May 2, 2001

1R22 Surveillance Testing

.1 Surveillance Testing of the Unit 1 West Residual Heat Removal Pump

ASME OMa-1988, Part 6	Inservice Testing of Pumps in Light Water Reactor Power Plants	
Unit 1 Technical Data Book, Figure 1-15.1	Safety Related Pump Inservice Testing Hydraulic Reference	Revision 69
12-EHP 5070.ISI.017R	Section XI Centrifugal Pump Performance Verification	Revision 7
ENSM 971016AF	RHR Deadheading	Revision 0
CR 01095067	Unit 1 west RHR pump differential pressure higher that Technical Data Book action level	April 5, 2001
CR 01129106	NRC questioned basis for residual heat removal inservice test action limits associated with pump deadheading	May 9, 2001

.2 Periodic Seal Injection Line Resistance Measurement

1-SGRP-009-N	D.C. Cook Unit 1 Tech Spec Review, B&W Replacement Steam Generators	Revision 0

1-SGRP-031-N	Best Estimate Flow & Irrecoverable Pressure Drop, B&W Replacement Steam Generators	Revision 0
1-SGRP-006-N	Cook Unit 1 OSG [Original Steam Generator]-RSG [Replacement Steam Generator] Comparison, B&W Replacement Steam Generators	Revision 0
HXP890724JJR	Controlled Leakage Tech Spec Calculation for AEP:NRC:1070	Revision 0
02-OHP 4030.STP.052L	Controlled Leakage Verification Test	Revision 4
01-OHP 4030.STP.052L	Controlled Leakage Verification Test	Revision 4
DIT B-01493	Incorporation of Measurement Uncertainties into the Seal Line Resistance Acceptance Criteria for Use in 01-OHP 4030.STP.052L	Revision 0
DIT B-01167	Incorporation of Measurement Uncertainties into the Seal Line Resistance Acceptance Criteria for Use in 01 and 02-OHP 4030.STP.052L	Revision 2
Letter AEP NRC 1070	Controlled Leakage Tech Spec Change	October 17, 1989
	NRC Safety Evaluation Report for Amendment Nos. 162 and 146 to Facility Operating License Nos. DPR-58 and DPR-74 (TAC Nos. M75243 and M75244),	February 13, 1992
DIT B-2032	Impact of Steam Generator Pressure Drop Change on Seal Line Resistance (R_{SL}) Calculation in Section 4.6 of Controlled Leakage Verification Test, OHP 4030-STP-052L, Units 1 and 2	Revision 0
CR 01124053	NRC questioned uncertainty analysis methodology for seal line resistance measurement test acceptance criteria	May 4, 2001
CR 01117020	NRC identified that the impact of Unit 1 steam generator replacement on seal line resistance technical specification was not reviewed	April 27, 2001

.3 Type C Containment Leak Rate Testing

01 EHP 4030.STP.203	Type B and C Leak Rate	Revision 4
ATR 1-CNTMT-1	Containment Systems - Containment Leakage	

.4 Surveillance Test of Unit 1Turbine Driven Auxiliary Feedwater Pump

Unit 1 TS 3.7.1.2	Auxiliary Feedwater System	
UFSAR Section 10.5.2	Auxiliary Feedwater System	
01-OHP.4030.STP.017T	Turbine Driven Auxiliary Feedwater System Test	Revision 15
CR 01130056	Mechanical overspeed trip lever did not trip the trip and throttle valve as expected during 017T	May 10, 2001

.5 Functional Test of Unit 1 Power Range Nuclear Instrument N-42

UFSAR Section 7.4	Nuclear Instrumentation	
Unit 1 TS 2.2	Reactor Trip System Instrumentation Setpoints	
Unit 1 TS 3.3.1	Reactor Trip System Instrumentation	
01-IHP 4030.SMP.131	Power Range Nuclear Instrumentation Functional Test and Calibration	Revision 0
JO R216219	Functional test of Unit 1 power range nuclear instrument N-42	May 9, 2001

4OA1 Performance Indicator Verification

PMP 7110.PIP.001	Regulatory Oversight Program Performance Indicators	Revision 0
PMI 7110	Regulatory Oversight Program	Revision 0
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	Revision 0

4OA3 Event Follow-Up

CR 98-1017

CR 97-2596	Engineered Safety Features actuation while filling steam generators to wet layup	September 23, 1997
01 OHP 4023.ES-0.1	Reactor Trip Response	Revision 14a
01 IHP 4030.SMP.115	Steam Generator Level Protection Set I Functional Test and Calibration	Revision 1
01 IHP 4030.SMP.116	Steam Generator Level Protection Set II Functional Test and Calibration	Revision 1
01 IHP 4030.SMP.117,	Steam Generator Level Protection Set III Functional Test and Calibration	Revision 1
01 IHP 4030.SMP.118,	Steam Generator Level Protection Set IV Functional Test and Calibration	Revision 1
Technical Specification Table 2.2-1	Reactor Trip System Instrumentation Trip Setpoints	
Technical Specification Table 3.3-4	Engineered Safety Feature Actuation System Instrumentation Trip Setpoints	

4OA5 Other

CR 99-19650	Perform a review of the design of the 600 VAC ungrounded system to document the insulation is rated for the system	July 27, 1999