July 24, 2002

Mr. J. Forbes
Site Vice-President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC INSPECTION REPORT 50-263/02-04(DRP)

Dear Mr. Forbes:

On June 30, 2002, the NRC completed an inspection at your Monticello Nuclear Generating Plant. The results of this inspection were discussed on July 2, 2002, with Mr. Purkis and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of activities conducted under your license as they relate to reactor safety, verification of performance indicators, event followup, occupational radiation safety, and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified two issues of very low safety significance (Green) and within the licensee response band. One Non-Cited Violation (NCV) of NRC regulations was identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/

Bruce L. Burgess, Chief
Branch 2
Division of Reactor Projects

Docket No. 50-263
License No. DPR22

Enclosure: Inspection Report 50-263/02-04(DRP)

See Attached Distribution

DOCUMENT NAME: C:\ADAMS\Cache\ML0220602460.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		RIII		RIII		RIII	
NAME	BBurgess:dtp							
DATE	07/ /02							

OFFICIAL RECORD COPY

cc w/encl: J. Purkis, Plant Manager
R. Anderson, Executive Vice President
 and Chief Nuclear Officer
Nuclear Asset Manager
Site Licensing Manager
Commissioner, Minnesota Department of Health
J. Silberg, Esquire
 Shaw, Pittman, Potts, and Trowbridge
R. Nelson, President
 Minnesota Environmental Control Citizens
 Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer
 Wright County Government Center
Commissioner, Minnesota Department of Commerce
P. Marker, Office of Attorney General

cc w/encl: J. Purkis, Plant Manager
R. Anderson, Executive Vice President
and Chief Nuclear Officer
Nuclear Asset Manager
Site Licensing Manager
Commissioner, Minnesota Department of Health
J. Silberg, Esquire
Shaw, Pittman, Potts, and Trowbridge
R. Nelson, President
Minnesota Environmental Control Citizens
Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer
Wright County Government Center
Commissioner, Minnesota Department of Commerce
P. Marker, Office of Attorney General

ADAMS Distribution:
WDR
DFT
SXM1
RidsNrrDipmIipb
GEG
HBC
SXB3
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/02-04(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	April 1 through June 30, 2002
Inspectors:	S. Burton, Senior Resident Inspector D. Kimble, Resident Inspector J. Neurauter, Regional Engineering Inspector M. Mitchell, Senior Radiation Specialist
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000263/02-04(DRP), Nuclear Management Company, LLC; 04/01-06/30/2002; Monticello Nuclear Generating Plant; Flood Protection Measures; Event Followup.

The inspection was conducted by resident inspectors, a regional engineering inspector, and a regional radiation protection inspector. The report covers a three month period of inspection. Two findings and one NCV were identified by inspectors. The significance of most findings is indicated by their color (green, white, yellow, red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "green," or may be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

Cornerstone: Mitigating Systems

- GREEN. Inspectors identified debris in the emergency core cooling system (ECCS) corner rooms which potentially could have had an adverse effect on installed flood protection equipment during an internal flooding event. The lack of adequate debris control procedures and instructions was determined to be a Non-Cited Violation (NCV) of Criterion V, 10 CFR 50, Appendix B.

  The finding was determined to be of very low safety significance and within the licensee's response band due to the very low risk associated with the event that was identified during a case specific Phase 3 SDP. (Section 1R06)

Cornerstone: Initiating Events

- GREEN. An undocumented modification performed on the Mechanical Pressure Regulator (MPR) rate feedback bellows in 1973 was determined to have contributed to the failure of the bellows causing a turbine load reject and reactor scram.

  This finding was determined to be of very low safety significance because of the age of the modification and all major plant equipment responded to the scram as designed. (Section 4OA3)

Report Details

Summary of Plant Status

The plant began the inspection period operating at full power. Power was reduced to approximately 90 percent on May 11, 2002 to permit quarterly turbine valve testing. Power was subsequently reduced to 75 percent on May 12, 2002, to facilitate main steam isolation valve (MSIV) testing. The plant returned to full power operation upon successful completion of MSIV testing on May 12. On June 28, 2002, operators were forced to rapidly reduce power to approximately 42 percent and enter single recirculation loop operation due to an oil leak on No. 12 recirculation motor-generator (MG) which required the MG to be secured (Section 4OA3.4). The plant returned to full power operation on June 29, 2002, following repairs to the No. 12 recirculation MG oil system, and remained at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors performed a walkdown of the licensee's preparations for adverse weather, including conditions that could lead to loss of off-site power and conditions that could result from high temperatures or high winds. The licensee's procedures and preparations for the impending tornado season were reviewed by the inspectors and were verified to be adequate. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed a partial walkdown of the following equipment trains to verify operability and proper equipment lineup. These systems were selected based upon risk significance, plant configuration, system work or testing, or inoperable or degraded conditions.

- Reactor Core Isolation Cooling (RCIC) During Feedwater Control System Maintenance

- Secondary Containment System

- Division I Low Pressure Coolant Injection (LPCI) with Division II Out-of-Service for Routine Maintenance

The inspectors verified the position of critical redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup.

Due to the system's risk significance, the inspectors selected the high pressure core injection (HPCI) system for a complete walkdown. The inspectors walked down the system to verify mechanical and electrical equipment lineups, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors walked down the following risk significant areas looking for any fire protection issues. The inspectors selected areas containing systems, structures, or components that the licensee identified as important to reactor safety.

- Fire Zone 1-E, HPCI Room

- Fire Zone 1-F, Torus Area (Elevation 896' and 923')

- Fire Zone 1-C, RCIC Room

- Fire Zone 3-B, Standby Liquid Control (SBLC) Room

- Fire Zone 5-B, Reactor Building (Elevation 1001')

- Fire Zone 5-A, Reactor Building (Elevation 1001')

- Fire Zone 23-A, Intake Structure Corridor

- Fire Zone 23-B, Intake Structure Pump Room

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, and barriers to fire propagation.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors selected the following plant areas for a semiannual review of internal flooding susceptibility and adequacy of flood protection features.

- Emergency Core Cooling System (ECCS) Corner Rooms
- Emergency Diesel Generator (EDG) Rooms

The areas were selected based upon the risk-significant systems, structures, and components (SSCs) present. Walkdowns and reviews performed considered unanalyzed sources for internal flooding, design measures, adequacy of barrier and penetration seals, drain systems and sumps, material condition including debris that could impact protective measures, penetration and room seals, performance and surveillance tests, procedural adequacy, and compensatory measures.

b. Findings

(1) Introduction

A finding of very low significance (Green) and an associated Non-Cited Violation (NCV) related to inadequate internal flood protection controls were identified by the inspectors.

(2) Description

During inspections of both the Division I and Division II ECCS corner rooms, the inspectors noted significant loose material and debris on the floors of the rooms. The material varied in size and quantity and consisted mainly of used anti-contamination clothing and rags, which were apparently being used to catch oil and/or other fluids dripping from mechanical components in the rooms. In addition, the inspectors noted that both ECCS corner rooms contained two metal trash cans of approximately 35 gallon capacity. One can in each room was being used as a receptacle for loose trash, while the other was being used to receive used anti-contamination clothing. The inspectors estimated that all four cans were between **b** and ¾ full.

Several scenarios related to internal flooding in the ECCS corner rooms can be postulated. While ECCS corner rooms each contain multiple floor drains, it is the two 100 gpm sump pumps in each room which are relied upon to mitigate postulated internal flooding events. The ECCS corner room floor drains are tied together with the torus room floor drain system, and hence, all three rooms are essentially connected. The worst-case postulated internal flooding event for the ECCS corner rooms actually occurs in the torus room, and floods the two ECCS corner rooms via the interconnected floor drain system. The licensee had previously modified the interconnecting floor drain system by installing orifices in the lines between the ECCS corner rooms and the torus room to limit flow into the ECCS corner rooms to within the capacity of the rooms' sump pumps.

Following discussions with the licensee's engineering staff, the inspectors concluded that the two worst-case scenarios involved either a break of the non-seismically qualified 18-inch service water header in the reactor building, or a break of the non-seismically qualified 12-inch condensate storage tank (CST) supply line in the reactor building. In the case of the service water header, the break could occur at any number of locations and/or elevations within the reactor building, and assuming all three service water pumps are available, the total flow rate available would be 19,500 gpm (6,500gpm x 3). In all instances, water flowing from the broken header eventually cascades to the lowest point in the reactor building, which is the torus room. Similarly, with the break of the CST supply line, the break occurs in the reactor building and water eventually cascades into the torus room. However, unlike the service water header break which results in pump-driven flooding, the postulated CST break is a gravity drain situation which can only be isolated by a large manually-operated valve and has the potential for 460,000 gallons of water to be available for internal flooding of the torus room.

The inspectors hypothesized that during a postulated internal flooding event, the loose material and debris on the floor of the ECCS corner rooms could be swept into the suctions of the rooms' sump pumps, potentially rendering these components inoperable. Likewise, the inspectors also postulated that the two 35 gallon cans in each room, which were freestanding and not secured in any fashion, could be toppled during this same postulated internal flooding event and provide additional material which could be swept into the suctions of the rooms' sump pumps.

With the sump pump suctions in both ECCS corner rooms clogged with the loose material and debris noted in each room, the inspectors hypothesized that a worst-case postulated internal flooding event in the torus room could flood out both ECCS corner rooms and render all four residual heat removal (RHR) pumps, both core spray pumps, and their supporting power-operated valves and instrumentation inoperable.

The licensee entered this issue into their corrective action program as condition report (CR) 20022314, “Loose Material In the Reactor Building and Turbine Building Could Adversely Impact Internal Flooding Protection.” Evaluation of this issue by the licensee revealed that the condition resulted from “inadequate housekeeping practices with regards to internal flooding protection.” Corrective actions recommended by the licensee included revisions to the housekeeping procedure and a review of other methods, procedures and practices, such as the guidelines for the use of anti-contamination clothing barrels, that may have an impact on the internal flooding analysis.

(3) <u>Analysis</u>

The inspectors evaluated the issue using NRC Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Disposition Screening." The inspectors determined that a performance deficiency existed with respect to compliance with internal flood control measures as a result of the amount and location of debris identified in the ECCS corner rooms during this inspection.

This issue was evaluated by the inspectors to determine whether or not it was of more than minor significance in accordance with the criteria of IMC 0612. As part of this effort, the inspectors focused on the objective of the mitigating systems cornerstone of reactor safety which is, "to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage)." It was determined that the finding was of more than minor significance in that it did affect the mitigating system cornerstone objective. Specifically, the lack of proper internal flood protection controls could directly impact the availability, reliability, and capability of all low pressure ECCS pumps located in the two ECCS corner rooms.

The inspectors analyzed this issue using the Significance Determination Process (SDP) contained in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." During a Phase 1 SDP review, the inspectors determined that a case-specific Phase 3 SDP analysis was required because the issue involved flooding, an area outside the capabilities of the site specific Phase 2 SDP. The inspectors forwarded information and documentation regarding this issue to the regional senior reactor analyst (SRA).

The SRA determined that the finding could impact several flooding initiating events (torus ring header, service water, and condensate system piping failures). In conjunction with the NRC evaluation of risk, the licensee's evaluation concluded that insufficient inventory exists in the CST to impact the ECCS corner rooms, therefore this flooding initiating event was eliminated from consideration. In determining the actual risk significance, the SRA reviewed the licensee's plant specific flooding analysis. The licensee's analysis concluded the issue was of very low risk significance (1.1E-8). This is based on a 2.5E-1 probability of both ECCS corner rooms flooding due to sump pump plugging, and a 2.5E-1 probability of personnel failing to detect and mitigate the plugging. While these assumptions were determined to be reasonable based on the ECCS corner room conditions at time of discovery, the licensee performed a sensitivity analysis to evaluate the uncertainty associated with plugging of the ECCS corner room sump pumps, and detection and mitigation of the plugging. This was found to be appropriate due to the high uncertainty associated with the plugging, detection, and mitigation. The licensee determined that the risk significance of the issue was in the range of E-7 to E-13, based on the sensitivity analysis. As a result, the SRA concluded the finding was of very low safety significance (Green) and within the licensee response band.

(4) <u>Enforcement</u>

The SSCs located within the ECCS corner rooms are safety related and are subject to the requirements of 10 CFR 50, Appendix B. 10 CFR 50, Appendix B, Criterion V, states, in part, that: "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances..." Contrary to this criterion, the licensee failed to establish procedures that were appropriate to the circumstances to control the amount of debris in the ECCS corner rooms. Specifically, the location and amount of debris in the ECCS corner rooms could adversely impact the capability of the installed flood protection equipment in the ECCS corner rooms during an internal flooding scenario. This is a violation of 10 CFR 50, Appendix B, Criterion V, however, because this issue was of very low safety significance and because it was entered into the corrective action program, the NRC is treating this issue as a Non-Cited Violation (NCV) consistent with Section VI.A.1 of the NRC's Enforcement Policy (NCV 50-263/02-04-01). The licensee has entered the issue into their corrective action program as Condition Report (CR) 20022314.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

During the week ending May 11, 2002, the inspectors observed the cleaning and inspection of the No. 12 reactor building closed cooling water (RBCCW) heat exchanger. The inspectors verified that the licensee was able to detect degraded conditions and checked that the licensee was adequately addressing problems which could result in an increase in plant risk. The inspectors reviewed the licensee's inspections and compared inspection results against the appropriate acceptance criteria, including the frequency of scheduled heat exchanger inspections being performed by the licensee.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

While at the licensee's training center during the week ending April 27, 2002, the inspectors observed a training crew during an evaluated simulator scenario. The scenario included a loss of non-vital 4160 Vac switchgear, a reactor scram, and an unisolable leak inside containment. The operators also responded to a loss of feedwater combined with a reactor power reduction, increasing containment pressure and lowering reactor vessel level, and a reactor depressurization to permit reactor vessel refill. Areas observed by the inspectors included: clarity and formality of communications, timeliness of actions, prioritization of activities, procedural adequacy and implementation, control board manipulations, managerial oversight, emergency plan execution, and group dynamics.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed and observed emergent work, preventive maintenance, or planning for risk significant maintenance activities. The inspectors observed maintenance or planning for the following activities undergoing scheduled or emergent maintenance:

- "A" Feedwater Regulating Valve (FRV) Troubleshooting and Repairs

- No. 11 EDG Emergency Service Water (ESW) Pump Replacement

- No. 12 EDG Governor Failure During Routine Surveillance Testing

The inspectors also reviewed the licensee's evaluation of plant risk, risk management, scheduling, and configuration control for these activities in coordination with other scheduled risk significant work. The inspectors verified that the licensee's control of activities considered assessment of baseline and cumulative risk, management of plant configuration, control of maintenance, and external impacts on risk. In-plant activities were reviewed to ensure that the risk assessment of maintenance or emergent work was complete and adequate, and that the assessment included an evaluation of external factors. Additionally, the inspectors verified that the licensee entered the appropriate risk category for the evolutions.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Routine Review of Licensee Operability Evaluations

a. Inspection Scope

The inspectors reviewed the technical adequacy of the following operability evaluations to determine the impact on Technical Specifications (TS), the significance of the evaluations, and to ensure that adequate justifications were documented.

- Use of Tape to Cover Opening on MCC [Motor Control Center] D312, CR 20023411

- Diesel Oil Transfer Pump Tripped While Filling the Diesel Fire Pump Fuel Oil Tank, CR 20025381

Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item (URI) 50-263/00-08-01: "Potential Vulnerability Associated With The RHRSW [Residual Heat Removal Service Water] Header Cross-tie"

The licensee clarified their position regarding a potential water hammer due to steam voiding following restart of the RHRSW system after a loss of off-site power (LOOP). The licensee developed enhancements to Procedure B.08.01.03-05 to initially restrict flow and condense any vapor cavity without creating a significant pressure surge. This resulted in the water hammer effects being considered negligible. Additionally, calculation CA-01-191 performed a qualitative comparison of steam voids and air voids and concluded that the dynamic effect of air voids was shown to be significantly less than steam voids. Because water hammer effects due to air voids in the RHRSW system were considered to be small (less than 10 psig pressure increase), a detailed B31.1 dynamic stress analysis of the RHRSW system was not required.

Based on Ops Man B.08.01.03-05, enhancements that will effectively prevent a steam cavity water hammer and the negligible effects of non-condensable air voids, URI 50-263/00-08-01 can be closed.

1R16 Operator Workarounds (OWA) (71111.16)

a. Inspection Scope

During the week ending April 13, 2002, the inspectors reviewed OWA 00-007, "Opening of MO-2032 or Closure Of its Breaker (B4211) with Reactor Temperature >212 Requires Initiation of a Compensatory Fire Watch." The inspectors reviewed the workaround focusing on the operators' ability to respond to alternate transients while performing duties as fire watch.

The inspectors also performed a semiannual review of the cumulative effects of OWAs. The inspectors reviewed the cumulative effects of workarounds on the reliability, availability, and potential for improper operation of the system. Additionally, reviews were conducted to determine if the workarounds could increase the possibility of an initiating event, affect multiple mitigating systems, or impact the operators' ability to respond to accidents or transients.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the following post-maintenance activities for review. Activities were selected based upon the structure, system, or component's ability to impact risk.

- Replacement of "A" Feedwater Regulating Valve Manual/Automatic Control Station

- Replacement of No. 11 EDG ESW Pump

- Troubleshooting and Repair of No. 12 EDG Woodward Governor

- Repair of Failed Diesel Fuel Oil Transfer Pump

- Repair of “A” Division ATWS [Anticipated Transient Without Scram] Power Supply Inverter

The inspectors verified by witnessing the test or reviewing the test data that post-maintenance testing activities were adequate for the above maintenance activities. The inspectors reviews included, but were not limited to, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, technical specification applicability, system restoration, and evaluation of test data. Also, the inspectors verified that maintenance and post-maintenance testing activities adequately ensured that the equipment met the licensing basis, TSs, and USAR design requirements.

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing (71111.22)</u>

a. <u>Inspection Scope</u>

The inspectors selected the following surveillance test activities for review. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the unit if the condition were left unresolved.

- Scram Discharge Volume Hi Level Scram Test and Calibration

- RCIC 10-Year Hydrostatic Pressure Test

- No. 11 EDG Operability Test with No. 12 EDG Out-of-Service

- Fire Pump Simulated Auto-Actuation and Capability Test

- 345 KV Substation (48VDC and 125 VDC) Battery Operability Check and Review of Division I and II 250 VDC Battery Surveillance Data

The inspectors observed the performance of surveillance testing activities, including reviews for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, impact of testing relative to performance indicator reporting, and evaluation of test data.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary modification "Temporary Signal Recorder for FRV 'A'," Jumper/Bypass No. 02-16.

The inspectors reviewed the safety screening, design documents, USAR, and applicable TSs to determine that the temporary modification was consistent with modification documents, drawings, and procedures. The inspectors also reviewed the post- installation test results to confirm that tests were satisfactory and the actual impact of the temporary modification on the permanent system and interfacing systems were adequately verified.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

While at the licensee's training center during the week ending April 27, 2002, the inspectors reviewed a simulator-based training evolution to evaluate drill conduct and the adequacy of the licensee's critique of performance to identify weaknesses and deficiencies. The inspectors selected simulator scenarios that the licensee had scheduled as providing input to the Drill/Exercise Performance Indicator. The inspectors observed, when applicable, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Observations were compared to the licensee's observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator reported statistics. The simulator scenario observed resulted in an unusual event and alert classifications.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns

a. Inspection Scope

During the week ending June 22, 2002, a regional radiation specialist inspector reviewed the radiological conditions of work areas within radiation areas and high radiation areas in the reactor, radwaste, and turbine buildings. Specifically, the inspector selected three work areas that required specific radiation work permits (RWP), reviewed the RWP and electronic dosimeter (ED) settings to verify conformity with current surveys and plant policies. The three RWPs were:

- RWP 73 Radiological Controlled Areas excluding Locked High Radiation Areas
- RWP 155 947Radwaste - Laundry Drain Tank and Pump Room
- RWP 159 1027 Reactor - RX Building Crane

The inspector attended pre-job briefs for these job activities, if required, and observed worker and radiation protection technician performance in the work areas to verify that workers were knowledgeable of RWP requirements, ED settings and licensee procedures for implementing good health physics practice.

In areas adjacent to locked high radiation areas (LHRA), the inspector performed walkdowns, verified area doserates, and reviewed licensee controls to determine if the controls (i.e., surveys, postings, and barricades) were adequate to meet 10 CFR Part 20 and Technical Specification requirements. The specific LHRAs were:

- Condenser Room
- Steam Tunnel
- In-core Detector Room
- Turbine Bioshield

The inspector walked down the spent fuel pool to verify that physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel pool were stored in accordance with the licensee's program requirements.

The inspector reviewed the licensee's internal dose assessment program for any actual internal exposure greater than 50 millirem Committed Effective Dose Equivalent, in the last year, to assess the adequacy of the licensee's internal dose assessment program.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

During the week ending June 22, 2002, the inspector selectively reviewed nine year 2001 and year to date 2002 condition reports (CR) that addressed access control deficiencies and radiation worker and radiation protection technician practices to verify that the licensee had effectively implemented the corrective action program. Additionally, the inspector reviewed the licensee's first quarter 2002 quarterly Chemistry and Radiation Protection Effectiveness Report and 2001 and 2002 Nuclear Assurance Observation Reports to assess the control program since the last inspection.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Job Site Inspections and ALARA Control

a. Inspection Scope

During the week ending June 22, 2002, a regional radiation specialist inspector selected a number of post-refueling outage high exposure or high radiation area work activities reviews to evaluate the licensee's use of ALARA controls for each activity. This evaluation included review of Job-In-Progress activities conducted by the ALARA staff and post job ALARA reports.

The inspector reviewed the licensee's use of thermoluminescent dosimeters (TLD) and ED systems to track occupational exposure. This review included a review of exposure tracking detail, report timeliness and exposure distribution to verify effectiveness of support to control collective exposures ALARA.

b. Findings

No findings of significance were identified.

.2 Source Term Reduction and Control

a. Inspection Scope

During the week ending June 22, 2002, using licensee records, the inspector reviewed historical trends and status of tracked plant source terms to determine if the licensee was making allowances or developing plans for expected changes in the source term, due to plant performance or plant primary chemistry. Additionally, the review was conducted to assess the licensee's source-term control strategy, including the use of permanent shielding. The inspector reviewed the last 12 month assessment period to assess the licensee's source-term reduction plans and priorities.

b. Findings

No findings of significance were identified.

.3 Radiation Worker Performance

a. Inspection Scope

During the week ending June 22, 2002, the inspector reviewed radiation worker performance on specific jobs described in Section 2OS1.1 to assess if workers have sufficient training and skill levels, demonstrate the ALARA philosophy in practice, and comply with established procedure.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

During the week ending June 22, 2002, the inspector reviewed recent Nuclear Oversight field observations and an audit performed in 2002, and 13 selected condition reports generated from the refueling outage in 2001 and year to date 2002, relative to the ALARA. In addition, the inspector reviewed the results of the Chemistry and Radiation Protection Self-Assessment reports to determine if the licensee adequately identified individual problems and trends, evaluated contributing causes and extent of condition, and developed corrective actions to prevent recurrence.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events, Barrier Integrity

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors verified the accuracy and completeness of the "Unplanned Scrams per 7000 Critical Hours" performance indicator data submitted by the licensee from April 1, 2001, through March 31, 2002. The inspectors reviewed data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the TS requirements, plant records, procedural reviews, and reactor coolant sample data.

b. Findings

No findings of significance were identified.

.2 Scrams with Loss of Normal Heat Removal

a. Inspection Scope

The inspectors verified the accuracy and completeness of the "Scrams with Loss of Normal Heat Removal" performance indicator data submitted by the licensee from April 1, 2001, through March 31, 2002. The inspectors reviewed data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the TS requirements, plant records, procedural reviews, and reactor coolant sample data.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Leak Rate

a. Inspection Scope

The inspectors verified the accuracy and completeness of the "Reactor Coolant System Identified Leak Rate" performance indicator data submitted by the licensee from April 1, 2001, through March 31, 2002. The inspectors reviewed data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the TS requirements, plant records, procedural reviews, and reactor coolant sample data.

b. Findings

There were no findings identified during this inspection.

4OA2 Identification and Resolution of Problems (71152)

Cornerstone: Mitigating Systems, Barrier integrity, and Initiating Events

.1 Semi-Annual Inspection of Internal Flooding Susceptibility - EDG Rooms

(1) Introduction

As part of the semi-annual internal flooding inspection (Section 1R06), the inspectors verified that the licensee had entered identified problems into their corrective action program. The inspectors selected licensee corrective actions related to internal flood protection measures in the EDG rooms for periodic review of the problem identification and resolution program per NRC Inspection Procedure (IP) 71152. Additionally, the inspectors verified that: 1) the licensee identified issues at an appropriate threshold, 2) that these issues were correctly entered in the corrective action program, and 3) that these issues were properly addressed for resolution.

(2) Description

As documented in Section 1R06 of this report, the inspectors conducted an inspection of the licensee's internal flood protection measure in the EDG rooms. Following a walkdown of the EDG rooms, the inspectors questioned the licensee regarding the testing and periodic maintenance of the three EDG drain line backwater check valves. One of the purposes for these three valves is to preclude a postulated internal flooding event in one EDG room from affecting the other EDG room via the common floor drain lines.

The licensee responded that the check valves had been inspected in 1995, and found to be heavily encrusted with silt and mud. Since that time, no further inspections had been performed and the licensee had no periodic testing requirements for the valves in place.

Following receipt of the inspectors' questions, the licensee decided to open and inspect all three EDG room backwater check valves. Upon inspection, one valve, NW-9, was found to be heavily silted with a significant percentage of the drain line upstream of the check valve blocked by silt and mud. Another valve, NW-8, displayed light silting, while the third valve, NW-7, appeared to be relatively free of any silt or mud.

a. Effectiveness of Problem Identification

i. Inspection Scope

The inspectors reviewed CR 20022403 and CR 20022543. The inspectors' review included verification that problem identification was complete, accurate, and timely, and that the consideration of extent of condition, generic implications, common cause, and previous occurrences were adequate.

ii. Issues

The licensee identified the 1995 silting of the EDG backwater check valves as a condition requiring periodic preventative maintenance. The inspectors' review of the issue indicated that the licensee had some intention of creating a periodic task to open, inspect, and clean the backwater check valves at some undefined periodicity.

b. Prioritization and Evaluation of Issues

i. Inspection Scope

The inspectors reviewed CR 20022403 and CR 20022543. The inspectors considered the licensee’s evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues.

ii. Issues

The inspectors determined that the licensee had taken no action, following the1995 inspection of the EDG floor drain line backwater check valves, to prevent recurrence of check valve silting. Although the licensee had created an action request to place the EDG floor drain line backwater check valves into a preventative maintenance program, the length of elapsed time, 7 years, suggests that this action had been overlooked. The fact that silting had recurred as identified in the licensee's most recent inspection in 2002 confirmed the need for these components to be placed into a periodic maintenance and/or testing program. However, because the most recent check valve inspection results indicated that the drain lines and backwater check valves would have functioned if called upon, a violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective was determined to be minor and not subject to formal enforcement action in accordance with the NRC Enforcement Policy.

c. Effectiveness of Corrective Actions

i. Inspection Scope

The inspectors reviewed multiple related condition reports to determine if the condition reports addressed generic implications and that the corrective actions were appropriately focused to correct the problem.

ii. Issues

While corrective actions related to the silting condition of the EDG backwater check valves appears to be adequate and focused on the apparent cause of the condition, the inspectors noted that the planned preventative maintenance will require future review to determine actual effectiveness of the licensee's corrective action.

.2 Control of Plant Permanent and Temporary Modifications

(1) Introduction

The inspectors identified several issues in the area of modification control. The inspectors communicated their observations with the licensee and CR 20024362, “Apparent Adverse Trend of Unauthorized Modification,” and CR 20025312, “Additional Results from Assessment of Trends in the Jumper Bypass and Modification Processes” were generated. The inspectors selected these condition reports for a periodic review of problem identification and resolution per IP 71152.

(2) Description

Subsequent to the December 2001 refueling outage, the inspectors identified six issues that related to the recognition, control, or implementation of both permanent and temporary plant modifications. Each of the examples is listed below.

- On January 21, 2002, a reactor scram occurred which was subsequently attributed to an uncontrolled permanent plant modification to the main turbine pressure regulator feedback bellows installed in 1973 (Section 4OA3.1). Although the modification occurred in 1973, this issue appeared to be an undocumented field change that adversely impacted plant operations.

- In April, 2002, the licensee identified a second historical modification to plant floor drains. Installed screens, found below the drain element, had the potential to cause premature plugging of floor drains during an event. Upon discovery, the licensee identified and removed all of the screens. The installation of the screen was a historical field change that potentially impacted plant design.

- In April, 2002, inspectors identified that duct tape was installed on an environmentally qualified switchgear spare breaker cubicle opening. The licensee determined that this condition was contrary to requirements and that an appropriate cover should be installed. After completion of the necessary alteration paperwork, the licensee installed a protective metal cover. During prior inspections, the inspectors had noted that identical style covers were installed on other switchgear. The inspectors noted that an alteration had to be prepared for each individual cover, and that a generic design change did not exist. Without a generic design change, appropriate documentation regarding the quality of the alteration did not exist and the inspectors questioned the acceptability of the prior alterations. The licensee determined that several pre-existing identical alterations had been installed without any supporting analysis. In 2001, the inspectors had identified tape installed on switch gear and CR 20011742, "Contrary to Industry Standard, Duct Tape Is Used In Some Spare MCC Cubicles for FME Control" was generated. Evaluation of the use of tape revealed that taping of the switch gear was contrary to procedure and electrical shop practices. Appropriate corrective action was taken to remove the duct tape from this and other switch gear.

- In February, 2002, the inspectors identified that an evaluation for a temporary modification to bypass certain turbine vibration instrumentation did not appear to adequately address statements in the operating procedure, nor require a temporary change to the procedure. The licensee evaluated the issue and a temporary procedure change was made.

- In April 2002, the inspectors identified a plastic tube, from an apparent catch basin, that exited from under a removable wall and was routed to a floor drain in the RCIC room. Subsequently, the inspectors found that the installation was not controlled as a temporary or permanent modification. The licensee had installed a plastic catch containment behind protective paneling to gather and route water seepage that occurred from an underground conduit when high ground water conditions existed. The installation took place during the refueling outage and appeared permanent in nature. Subsequent to the inspectors observations, a procedure was being developed to control the use of these temporary alterations to the plant.

- In April 2002, the inspectors identified that test equipment installed for monitoring feedwater regulating valve performance was to be installed for greater than 24 hours and did not have temporary modification controls in place. The licensee issued a CR to evaluate this issue because it appeared to be contrary to procedural requirements.

Although minor in nature, each of the issues appeared to be contrary to procedural requirements. The inspectors noted that some of the issues were historical in nature, but they were included as examples because the issues appeared to be similar to recent observations.

a. <u>Effectiveness of Problem Identification</u>

i. <u>Inspection Scope</u>

The inspectors reviewed CR 20024362 and CR 20025312. The inspectors' review considered if problem identification was complete, accurate, and timely, and that the consideration of extent of condition review, generic implications, common cause, and previous occurrences were adequate.

ii. <u>Issues</u>

The inspectors reviewed CR 20024362, “Apparent Adverse Trend of Unauthorized Plant Modifications.” Although the condition report title indicated the scope of review was unauthorized plant modifications, the condition reports initially examined only those examples that raised the question of a possible trend. Subsequently, the licensee initiated CR 20025312, "Additional Results Form Assessment of Trends in the Jumper Bypass and

Modification Processes." This second condition report recognized that CR 20024362 was narrowly focused and attributed the cause to mis-communications.

CR 20024362 also identified that there was "no adverse trend," but indicated that "the problems when taken together indicated a need for additional corrective action" (i.e., a trend). Additionally, CR 20025312 identified that the scope of the issue extended to additional condition reports not initially considered in the original condition report, CR 20024362. CR 20025312, indicated that the narrowly focused extent of condition of CR 20024362 was due to a communications error between plant staff and management. CR 20025312 noted that some of the newly identified examples were being reviewed for trends specific to the issue, for example Part 50.59 inconsistencies. CR 20025312 identified additional examples or related condition reports that were not considered in the original assessment. CR 20025312 concluded that the "additional examples of unauthorized modifications do not affect the conclusions or recommendations of CR 20024362."

The inspectors independently reviewed condition reports for common cause issues and previous occurrences. Only a minimal number of similar conditions that were not included in the extent of condition review for CR 20024362 and CR 20025312 were identified. One example of a related condition report that was not captured by the licensee's review was CR 20021422, "Multiple Site Performance Errors Associated With The Modification Process Indicates Adverse Trend." This condition report was issued at the request of both the plant manager and site vice president in February 2002, and closed in March 2002. The extent of condition in CR 20021422 was limited to the items cited in the condition report as the indicator of a possible trend. Additionally, the assessment appeared to address only the concerns raised by plant management.

b. Prioritization and Evaluation of Issues

i. Inspection Scope

The inspectors reviewed CR 20024362 and CR 20025312. The inspectors considered the licensee's evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues.

ii. Issues

Corrective actions appear to be based upon a qualitative assessment of risk. The inspectors found no examples where the licensee improperly implemented their procedure for categorizing risk issues. The licensee's procedure used a table of examples and guidelines to determine report classification. The procedure does not appear to contain a formal mechanism to consider the risk significance of an issue. For the issues reviewed, the inspectors did not find any discrepancies from what would be the apparent risk and the categorization assigned by the licensee.

The inspectors considered the effectiveness of operability determinations in their review of related condition reports. In general, there were few operability evaluation required for the condition reports reviewed. However, one example of an incomplete operability determination was associated with the proposed installation of a redesigned spray hood for emergency service water pumps, CR 20021002. The initial modification was approached from a fire protection aspect, but the impact of the maintenance on this design feature was not considered when operability was reviewed. This issue was identified by the resident inspectors and corrected before the modification was implemented. Additionally, upon review of CR 20023333, "Catch Basin Placed Inside Conduit Panel Using GRAMA [General Repair and Maintenance Activity] Work Order Located In The RCIC Room," the inspectors identified that the operability evaluation did not document certain engineering aspects of the evaluation performed. For example, the evaluation indicated that a walk-down of the condition was performed and that there were no operability concerns. The condition report indicated that the walk-down focused on seismic concerns, but did not address any aspects of flooding or spraying that could occur as a result of temporary installation.

c. Effectiveness of Corrective Actions

i. Inspection Scope

The inspectors reviewed multiple related condition reports to determine if the condition reports addressed generic implications and that corrective actions were appropriately focused to correct the problem.

ii. Issues

The inspectors reviewed multiple condition reports identified through independent searches and referenced in CR 20024362 and CR 20025312. Corrective actions related to each condition report appeared to be adequate and were focused on the apparent cause of each condition. The corrective actions from CR 20024362 appear to be training based and focused on the recognition of unauthorized modifications. Additionally, the inspectors noted that the proposed training-based corrective action may be sufficient to preclude recurrence. However, future review will be required to determine if the corrective actions were narrowly focused.

.3 Reactor Building Railroad Airlock Doors

As documented in Section 4OA7 of this report, the licensee identified a violation of very low safety significance (Green) associated with their ineffective evaluation and corrective action for several occurrences of the inner railroad airlock doors being left open and unattended contrary to posted instructions.

4OA3 Event Follow-up (71153)

Cornerstones: Initiating Events and Mitigating Systems

.1 (Closed) Licensee Event Report (LER) 50-263/2002-01, Revisions 0 and 1: "Mechanical Pressure Regulator (MPR) Failure Causes Reactor Scram"

a. Inspection Scope

During the week ending April 13, 2002, inspectors evaluated LER 50-263/2002-01, Revision 0, "Mechanical Pressure Regulator Failure Causes Reactor Scram." Revision 1 of this LER was subsequently reviewed by inspectors during the week ending May 25, 2002.

b. Findings

(1) Introduction

A finding of very low significance (Green) related to an undocumented modification made to the MPR which contributed to a recent scram was identified by the inspectors.

(2) Description

On January 21, 2002, the unit experienced a reactor scram from a turbine load reject signal while operating at full power. A review of plant data associated with the scram led the licensee to conclude that the MPR in the plant's steam pressure control system had been the cause of the scram. Further investigation by the licensee revealed that the MPR rate feedback bellows had failed, and that the bellows had been physically modified by the attachment of several solid bars which had been soldered to the bellows to adjust its feedback characteristics. This modification to the bellows had apparently been performed in 1973, with little or no documentation, and contributed to the bellows failure. Prior to unit restart, the licensee replaced the MPR rate feedback bellows with a new, unmodified component pursuant to the manufacturer's original specifications.

(3) Analysis

Inspectors determined the licensee's performance to be deficient in the area of properly documenting a design change to the plant. Specifically, the licensee failed to adequately document the modification of the MPR rate feedback bellows in 1973.

Inspectors determined the finding had the potential to be more than minor in that the undocumented modification to the MPR rate feedback bellows had an actual and credible impact on plant safety. Specifically, the undocumented modification contributed to the bellows failure and the resulting reactor scram. Consideration was also given the to amount of time between the undocumented modification and the plant scram. Additionally, inspectors determined that the issue affected the initiating events cornerstone of reactor safety in that it did cause or increased the frequency of an initiating event, in this case, a reactor scram.

The inspectors employed the SDP to determine the potential risk significance of the finding. During a Phase 1 SDP review, the inspectors determined that because all major plant equipment responded to the scram as designed and no mitigating systems were lost, the finding was determined to be of very low significance and within the licensee's response band (Green). The licensee had entered this issue into their corrective action program as CR 20020457 and CR 20020573.

.2 (Closed) Licensee Event Report (LER) 50-263/2002-02, Revisions 0 and 1: "Application of Instrument Deviation Acceptance Criteria Allowed As-Found Settings to be Outside Technical Specification Value"

a. Inspection Scope

During the week ending May 25, 2002, inspectors evaluated LER 50-263/2002-02, Revisions 0 and 1, "Application of Instrument Deviation Acceptance Criteria Allowed As-Found Settings to be Outside Technical Specification Value."

b. Findings

While monitoring a periodic TS instrumentation calibration procedure, inspectors noted that the licensee's as-found acceptance criteria for several instrument parameters was not consistent with the specified TS limit. Upon further review, the inspectors identified that the licensee was applying a deviation given in the TS bases to as-found TS limits for most instrumentation. This allowed deviation encompassed the maximum uncertainty and drift for a given as-found instrument parameter, which when applied to the specified TS limit provided the acceptable range of as-found instrument parameter values. Any as-found value beyond this constituted an inoperable instrument. While this was a historic practice associated with the licensee's custom Technical Specifications and TS-related instrumentation surveillances, the inspectors questioned the practice as it allowed for the possibility that an as-found value outside the TS limit yet within the acceptable range created by use of the TS bases deviation to be considered operable while, strictly speaking, in violation of the TS limit.

Following receipt of the inspectors' concerns about this practice, the licensee reviewed past TS instrumentation surveillances to determine whether or not any TS limits had been violated by actual as-found instrument parameter values. The licensee found that on at least two occasions, as-found trip values for multiple low condenser vacuum scram instruments had been beyond the specified TS instrument limit, yet the instruments were not declared inoperable and no applicable limiting condition for operation was entered due to the as-found values being within the acceptable range created by adding the TS bases deviation to the TS limit. The licensee evaluated these instances to be of very low safety significance since even though the plant was being operated with multiple channels of low condenser vacuum scram instrumentation inoperable per the specified TS parameter limit, all channels of instrumentation were capable of performing their safety functions at all times. This was due to the fact that the as-found parameter values for these instruments were all within the acceptance band created by using the TS bases deviation.

The inspectors determined that no findings of significance were associated with this event. The licensee had entered this issue into their corrective action program as CR 20021013.

.3 (Closed) Licensee Event Report (LER) 50-263/2002-03: "Primary Containment Group 3 Isolation Signal on High Flow While Repressurizing Reactor Water Cleanup System Piping"

a. Inspection Scope

During the week ending June 29, 2002, inspectors evaluated LER 50-263/2002-03, "Primary Containment Group 3 Isolation Signal on High Flow While Repressurizing Reactor Water Cleanup System Piping."

b. Findings

During full power operation on March 22, 2002, an automatic Group 3 containment isolation signal was generated on high reactor water cleanup (RWCU) system flow. This signal resulted in the automatic closure of containment isolation valves in the RWCU system and sample connections from the recirculation system. While restoring part of the RWCU system to service following maintenance, operations personnel failed to fully fill and vent the isolated portion of the system. Consequently, upon restoration high RWCU flow was experienced as water rushed into the previously isolated portion of the system to fill voids which had developed. Following an investigation of the isolation event, the licensee successfully filled, vented, and repressurized the isolated portion of the RWCU system and returned it to service.

The inspectors determined that no findings of significance were associated with this event. The licensee had entered this issue into their corrective action program as CR 20022889.

.4 Reactor Single Loop Operation

a. Inspection Scope

On June 28, 2002, The inspectors observed entry into and recovery from single loop operations. Single loop operations resulted from a failure of a mechanical connection on the No. 12 recirculation motor generator (MG) lubricating oil pump discharge oil filter. The inspectors observed control room operations, procedure usage, plant parameters, alarms and conditions related to the event, resolution of the condition, and the licensee's management of risk.

b. Findings

No findings of significance were identified.

4OA6 Meeting

Exit Meeting

The inspectors presented the inspection results to Mr. Purkis and other members of licensee management on July 2, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violation

Cornerstone: Barrier Integrity

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Criterion XVI of 10 CFR 50, Appendix B, requires that nonconformances are promptly identified and corrected. Following several instances where the resident inspectors identified that the reactor building railroad airlock inner doors were found open and unattended contrary to posted instructions, the licensee evaluated the series of occurrences and determined that their repeated nature was due to ineffective evaluation and corrective action associated with the initial condition report. However, because in each case the railroad airlock outer doors were closed, secondary containment integrity was never lost and the violation is not more than of very low significance, within the licensee's response band, and is being treated as a NCV. The licensee entered this issue into their corrective action program as CR 20024417.

KEY POINTS OF CONTACT

Licensee

G. Bregg, Manager, Quality Services
R. Deopere, Inservice Inspection Supervisor
D. Fadel, Director of Engineering
J. Forbes, Site Vice-President
J. Grubb, Operations Manager
K. Jepson, Radiation Protection and Chemistry Manager
B. Linde, Security Manager
D. Neve, Licensing Project Manager
J. Purkis, Plant Manager
B. Sawatzke, Maintenance Manager
C. Schibonski, Safety Assessment Manager
E. Sopkin, Engineering Manager

NRC

B. Burgess

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-263/02-04-01	NCV	Inadequate Debris Control in ECCS Corner Rooms Challenges Internal Flooding Analysis (Section 1R06)

Closed

50-263/02-04-01	NCV	Inadequate Debris Control in ECCS Corner Rooms Challenges Internal Flooding Analysis (Section 1R06)
50-263/02-04-02	NCV	Modification to MPR Without 10 CFR 50.59 Evaluation Contributes to Reactor Scram (Section 4OA3.1)
50-263/00-08-01	URI	Potential Vulnerability Associated With The RHRSW [Residual Heat Removal Service Water] Header Cross-tie (Section 1R15)
50-263/2002-01, Revisions 0 and 1	LER	Mechanical Pressure Regulator (MPR) Failure Causes Reactor Scram (Section 4OA3.1)
50-263/2002-02, Revisions 0 and 1	LER	Application of Instrument Deviation Acceptance Criteria Allowed As-Found Settings to be Outside Technical Specification Value (Section 4OA3.2)
50-263/2002-03	LER	Primary Containment Group 3 Isolation Signal on High Flow While Repressurizing Reactor Water Cleanup System Piping (Section 4OA3.3)

Discussed

None

LIST OF ACRONYMS USED

ALARA As-Low-As-Is-Reasonably-Achievable
ASME American Society of Mechanical Engineers
ATWS Anticipated Transient Without Scram
AWI Administrative Work Instruction
CFR Code of Federal Requirements
CR Condition Report
CST Condensate Storage Tank
DBD Design Basis Document
DC Direct Current
DRP Division of Reactor Projects
ECCS Emergency Core Cooling System
ED Electronic Dosimeter
EDG Emergency Diesel Generator
ESW Emergency Service Water
EWI Engineering Work Instruction
FFD Fitness For Duty
FOI Follow-On Item
FRV Feedwater Regulating Valve
FSAR Final Safety Analysis Report
HPCI High Pressure Core Injection
IMC Inspection Manual Chapter
IP Inspection Procedure
IPE Individual Plant Examination
IPEEE Individual Plant Examination of External Events
IR Inspection Report
kV Kilovolt
LER Licensee Event Report
LHRA Locked High Radiation Area
LOCA Loss of Coolant Accident
LOOP Loss of Off-site Power
LPCI Low Pressure Coolant Injection
MG Motor-Generator
MPR Mechanical Pressure Regulator
MSIV Main Steam Isolation Valve
NCV Non-Cited Violation
NEI Nuclear Energy Institute
NIOSH National Institute of Safety & Health
NMC Nuclear Management Company
NUMARC Nuclear Management and Resources Council
OWA Operator Workaround
OWI Operations Work Instruction
PI Performance Indicator
PM Planned or Preventative Maintenance
psig Pounds Per Square Inch Gauge
QA Quality Assurance

RBCCW	Reactor Building Closed Cooling Water
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RP	Radiation Protection
RPS	Radiation Protection Specialist
RPS	Reactor Protection System
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SBLC	Standby Liquid Control
SCR	Screening
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRI	Safety Review Item
SSC	Systems, Structures, and Components
SWI	Scheduling Work Instruction
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current
WO	Work Order

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather

CR 20003476	Ensuring Continued Safe Plant Operation and Minimizing Requests for Enforcement Discretion During Extreme Weather Conditions	
2.3	USAR - Section 2 Site and Environs, Metrology	Revision 18
1150	Summer Checklist	Revision 31
1444	Post Severe Weather Checklist	Revision 3
APED-5696	Tornado Protection for Spent Fuel Storage Pool	November 1968
2000-15	NRC Regulatory Issue Summary 2000-15 Recommendations for Ensuring Continued Safe Plant Operation and Minimizing Requests for Enforcement Discretion During Extreme Weather Conditions	September 7, 2000

1R04 Equipment Alignment

	Drawings and Prints:	
M-125	- RCIC (Steam Side)	Revision AM
M-126	- RCIC (Water Side)	Revision AA
M-120	- Residual Heat Removal System (Division II)	Revision BH
M-121	- Residual Heat Removal System (Division I)	Revision BL
NF-36298-1	- Electrical Load Flow Diagram	Revision N
NF-36298-2	- DC Electrical Load Distribution Diagram	Revision A
NH-36249	- HPCI (Steam Side)	Revision AJ
NH-36249-1	- HPCI (Hydraulic Control and Lubrication System)	Revision C
NH-36250	- HPCI (Water Side)	Revision AB
	Operations Manual:	
B.2.3	- Reactor Core Isolation Cooling	
B.4.2	- Secondary Containment/Standby Gas Treatment Systems	
B.3.4	- Residual Heat Removal System	
B.3.2	- HPCI	Revision 4
	USAR:	Revision 18
Section 5.3	- Secondary Containment System	
Section 6.2	- Emergency Core Cooling Systems	
Section 6.2.4	- High Pressure Coolant Injection System (HPCI)	Revision 18
LER 95-05	Positive Pressure in Reactor Building During Ventilation Restoration	Revision 0

DBD B3.2	Design Basis Document - High Pressure Coolant Injection System	Revision 3
AG 1996-O-2	QA Internal Audit Report on Plant Operations	August 23, 1996
CR 20021047	Door-45 and Door-46 Have Multiple Signs Attached Including "CAUTION" Signs with Instructions	
CR 20020510	Rx Building Inner RR Door No 46 Found Open When No Work Was in Progress. This is Not Allowed Per the Caution Sign on the Door	
CR 20016321	Rx Building Railroad Door (Inner) Found Open With No Work in Progress and Unattended, Contrary to Posted Signs	
CR 20024899	Document in Dochandler the Position That Both Reactor Building Doors Need Not Be Closed to Meet Secondary Containment Design Basis	
	Procedures and Forms:	
0151-01	- Secondary Containment Capability Test	Revision 7
CR 20025514	Inadequate Guidance Provided for Monitoring HPCI Oil Resevoir Level Results in Questionable Oil Level	
CR 20025502	HPCI and RCIC Isometric Drawings Not Updated to Reflect that Startup Suction Strainers Have Been Removed	
B.4.2	Design Basis Document - Secondary Containment/Standby Gas Treatment Systems	Revision 2
2154-10	- High Pressure Coolant Injection System Prestart Valve Checklist	Revision 26
NX-8292-42	Byron-Jackson Vendor Manual	
NX-8292-54	Terry Turbine Vendor Manual	

<u>1R05</u> <u>Fire Protection</u>

NX-16991	Technical Manual, “Monticello Updated Fire Hazards Analysis”	
USAR 10.3.1	Fire Protection System	Revision 18

	Monticello Fire Strategies:	
A.3-01-E	HPCI Room	Revision 4
A.3-01-C	RCIC Room	Revision 2
A.3-01-F	Torus Area, Elevation 896' and 923'	Revision 4
A.3-03-B	Standby Liquid Control Area	Revision 6
	Procedures and Forms:	
0271	- “Fire Hose Station and Yard Hydrant Hose House Equipment Inspection”	Revision 27
0275-2	- “Fire Barrier Wall, Damper, and Floor Inspection”	Revision 18
0275-1	- “Fire Barrier Penetration Seal Visual Inspection”	Revision 9
0275-3	- Fire Door Inspections	Revision 23

<u>1R06</u> <u>Flood Protection Measures</u>

	Design Basis Documents:	
T.5	- External Flooding	Revision 3
T.8	- Internal Flooding	Revision 2
NSPLMI-95001	Individual Plant Examination of External Events (IPEEE)	Revision 1
NSPNAD-92003	Individual Plant Examination (IPE)	Revision 0
	USAR:	Revision 18
Section 12.2.1.7.1	- External Flooding	
Section 12.2.1.7.2	- Internal Flooding	
Section 2.4.1	- Surface Water	
Section 1.3.1.4	- Hydrology	
	Operations Manual:	
A.6	- Acts of Nature	Revision 13*
B.7.1	- Liquid Radwaste	
	Procedures:	
1478	- Annual Flood Surveillance	Revision 0
4AWI-04.02.01	- Housekeeping	Revision 6
1252	- RHR Pump Room Sump Pump Surveillance	Revision 6
1306	- Portable Diesel Oil Pump Operability Test	Revision 8
CR 20022314	Loose Materials in the Reactor Building and Turbine Building Could Adversely Impact Internal Flooding Protection	
CR 20022403	The EDG Room Backwater Check Valves Have Not been Inspected and Cleaned Since 1995. Corrective Actions For Prior Issue Not Completed	

NRC IN 83-44, Supplement 1	Potential Damage to Redundant Safety Equipment as a Result of Backflow Through the Equipment and Floor Drain System	August 30, 1990

1R07 Heat Sink Performance

	Procedures and Forms:	
4131-PM	- RBCCW Heat Exchanger Inspection and Cleaning	Revision 8
8151	- Heavy Load Movement Procedure	Revision 7
WO 0201183	PM 4131 No. 12 RBCCW Heat Exchanger	
B.2.5	Operations Manual: Reactor Building Closed Cooling Water System	
Section 10.4.3	USAR: Reactor Building Closed Cooling Water System	Revision 18
M-111	Drawing: Reactor Building Closed Cooling Water System	Revision AD

1R11 Licensed Operator Requalification Program

RQ-SS-38E	Loss of High Pressure Injection with Small LOCA	Revision 0
C.5-1100	Reactor Pressure Vessel Control	Revision 8
C.5-1200	Primary Containment Control	Revision 11
C.5-2002	Emergency Reactor Pressure Vessel Depressurization	Revision 5
C.4-G	Inadvertent ECCS Initiation	Revision 2
C.4-A	Reactor Scram	Revision 19
C.4-B.06.05.A	Feedwater Pump Trip	Revision 5

1R13 Maintenance Risk Assessments and Emergent Work Control

	10 CFR 50.59 Screenings:	
SCR 02-180	- Temporary Signal Monitoring for FRVs	Revision 0
SCR 02-172	- Manual Control of FRVs From Main Control Room	Revision 0
	Drawings:	
NF-36962	- External Connection Diagram - Reactor Control Bench Board C-05	Revision AD
NX-7828-26-2	- Connection Diagram - Panel 9-5	Revision Y
NX-7828-26-3	- Connection Diagram - Panel 9-5	Revision N

	Procedures and Forms:	
4AWI-04.04.03	- Bypass Control	Revision 17
4AWI-04.05.15	- Control of Troubleshooting Activities	Revision 0
C.6-CFW508	- Alarm Response - DFCS FWCV Demand Deviation	Revision 0
B.05.07-05	- System Operation - Reactor Level Control	Vol F - 2049
3704	- Determination/Re-termination Documentation Sheet	Revision 0
3063-05	- ASME Section XI Repair/Replacement Plan	Revision 6*
3108	- Pump/Valve/Instrument Record of Corrective Action	Revision 8
4109-01-PM	- EDG 8 Cycle Maintenance	Revision 3
3186-G-01-03	- QC Inspection Record for WO 0202336	Revision 5
3632	- Pocket Torquing Guide	Revision 1
	USAR:	
Appendix A	- Seismic Design Criteria	Revision 13
Section 8.4.1	- Safeguards EDG Systems	Revision 18
J/B 02-16	Temporary Signal Recorder for FRV 'A'	Revision 0
	Tagouts/Isolations:	Version 1
01-09337	- No. 11 EDG ESW Pump Replacement	
02-02335	- Troubleshoot/Repair No. 12 EDG Governor	
B.31.1	USAS/ASME - Power Piping	1967 Edition
WO 9404088	Engine Speed Drifts While Loaded	
WO 0109337	Replace No. 11 EDG ESW Pump; ?P in Alert Range	
WO 0201965	Investigate/Repair/Replace Regulating Valve 6-12A M/A Station	
WO 0201982	Investigate/Repair 'A' FRV M/A Station Control Loop	
WO 0202336	Replacement of No. 12 EDG Governor	
CR 20024045	#12 EDG Had to Be Manually Adjusted to Maintain 2500KW While Performing the Quarterly Operability Test 0187-02	
CR 20025459	Found Some Yoke to Bonnet Studs Loose on No. 13 and NO. 14 RHRSW Pump Discharge Valves	
CR 20023604	Questionable Mechanical Torquing Practices Associated with No. 11 EDG-ESW Pump Replacement Activity	

1R15 Operability Evaluations

CR 20023411	Tape on MCC D312	
CR 20025381	Diesel Oil Transfer Pump Tripped While Filling The Diesel Fire Pump Oil Tank	
CR 20025391	Tech Spec 3.13 Entry Requirements Unclear with Respect to Inoperable Components, Not Controls, Such as P-11 Oil Pump	
SRI 91-10	USAR Update Concerning Diesel Oil Day Tank	
SRI 91-27	Resident Inspector Identified Fire Issues for Diesel Generator Fuel Oil Systems	
SRI 90-037	Acceptability of the Diesel Fuel Oil Transfer House as a Class II Structure Housing Safety Related Class I Equipment	
SRI 91-034	Clarification of Diesel Oil Transfer System Redundancy Criteria	
	Internal Correspondence, R. A. Anderson to Matt Antony dated May 29, 2002, Subject: NRC Follow-On Questions Regarding Resolution of SSDI URI 50-263/00-08-01, Review of Potential Water Hammer in the RHRSW System	May 29, 2002
CR 20014565	Assess Potential Voiding of RHRSW during S/D Cooling with Subsequent Loss of Offsite Power	
CA-01-191	Evaluation of Potential Water Hammer Events within the RHR Service Water System, Approval Date 10-8-01	Revision 0
B.08.01.03-05	RHR Service Water System, System Operation	Revision 19

1R16 Operator Workarounds

	Monticello Operational Challenges List	April 11, 2002
	Monticello Operational Challenges History List	April 11, 2002
	Monticello Operational Challenges List: Acceptable As-is Report	April 11, 2002
	Operations Department Quarterly Effectiveness Reports - 4th Quarter 2001	March 6, 2002

	Operator Work-Around/Non-Transient Operator Work-Around Impact Factor Report	April 11, 2002
OWI-01.07	Operations Department Self Assessment	Revision 16
2220	Operational Challenge Resolution - Operator Workarounds for Item 00-007	Revision 3
CR 20023509	Operations Challenge Methodology to Address the Cumulative Affects of Operator Work-arounds During Plant Transients Questioned	
CR 20023506	Operational Challenges Determined to Be Acceptable As-is Are Not Tracked as Part of Aggregate Impact Factor	

<u>1R19</u> <u>Post-Maintenance Testing</u>

	10 CFR 50.59 Screenings:	
SCR 02-180	- Temporary Signal Monitoring for FRVs	Revision 0
SCR 02-172	- Manual Control of FRVs From Main Control Room	Revision 0
	Drawings:	
NF-36962	- External Connection Diagram - Reactor Control Bench Board C-05	Revision AD
NX-7828-26-2	- Connection Diagram - Panel 9-5	Revision Y
NX-7828-26-3	- Connection Diagram - Panel 9-5	Revision N
	Procedures and Forms:	
4AWI-04.04.03	- Bypass Control	Revision 17
4AWI-04.05.15	- Control of Troubleshooting Activities	Revision 0
C.6-CFW508	- Alarm Response - DFCS FWCV Demand Deviation	Revision 0
B.05.07-05	- System Operation - Reactor Level Control	Vol F - 2049
3704	- Determination/Re-termination Documentation Sheet	Revision 0
3063-05	- ASME Section XI Repair/Replacement Plan	Revision 6*
3108	- Pump/Valve/Instrument Record of Corrective Action	Revision 8
4109-01-PM	- EDG Cycle 8 Maintenance	Revision 3
3186-G-01-03	- QC Inspection Record for WO 0202336	Revision 5
4107-02-OCD	- 12 EDG 2 Cycle	Revision 3
0187-02	- 12 EDG and 12 ESW Pump System Tests	Revision 38
3069	- Post-Maintenance Activities Control Cover Sheet	Revision 11

	Tagouts/Isolations:	Version 1
01-09337	- No. 11 EDG ESW Pump Replacement	
02-02335	- Troubleshoot/Repair No. 12 EDG Governor	
02-80070	- 12 EDG 2 Cycle	
J/B 02-16	Temporary Signal Recorder for FRV 'A'	Revision 0
WO 0109337	Replace No. 11 EDG ESW Pump; ?P in Alert Range	
WO 0201965	Investigate/Repair/Replace Regulating Valve 6-12A M/A Station	
WO 0201982	Investigate/Repair 'A' FRV M/A Station Control Loop	
WO 0202336	Replacement of No. 12 EDG Governor	
WO 0202351	12 EDG High Speed Limit Switch Adjustment and Speed Control Band Setting	
CR 20025381	Diesel Oil Transfer Pump Tripped While Filling The Diesel Fire Pump Oil Tank	
3069	Post-Maintenance Testing Activities Control Cover Sheet for Work Order 0202930	Revision 11
CR 20025820	Fault IN “A” ATWS Inverter Required Removal From Service	
WO 0203125	Investigate/Repair ATWS Invt-A	
3069	Post-Maintenance Testing Activities Control Cover Sheet for Work Order 0203125	

1R22 Surveillance Testing

	Technical Specifications and Bases:	
3/4.1	- Reactor Protection System	
3/4.9.B.3	- Standby Diesel Generators	
	Drawings and Prints:	
M-119	- Control Rod Hydraulic System	Revision S
M-125	- RCIC (Steam Side)	Revision AM
M-126	- RCIC (Water Side)	Revision AA
M-133	- Diesel Oil System	Revision AD
Section XI	ANSI/ASME Boiler & Pressure Vessel Code - Rules for Inservice Inspection of Nuclear Power Plant Components	1986 Edition

CR 20023642	MO-2076 Failed to Fully Open During Step 41 of Test 0062, RCIC Steam Line High Area Temperature Test and Calibration	
CR 20023698	Special Guidance on Operation of MO-2076 Not in Place. NRC Resident Inspector Identified	
IEEE Standard 450-1980	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications	
NX-16647	C&D Battery Manual	
	Procedures and Forms:	
0187-01	- 11 EDG and 11 ESW System Tests	Revision 38
0255-08-IIB	- RCIC System Pressure Tests	Revision 5
0255-08-IA-1	- RCIC System Pump Flow and Valve Test	Revision 51
0006	- Scram Discharge Volume Hi Level Scram Test and Calibration Procedure	Revision 20
0266	Fire Pumps Simulated Auto-Actuation and Capability Test	Revision 37
4510-PM	Maintenance of On-Site Batteries and Battery Chargers at Monticello Nuclear Plant	Revision 13
0194	11 and 12 125 VDC Battery Operability Check	Revision 13
0185	345 KV Substation (48 VDC and 125 VDC) Battery Operability Check	Revision 11
0193-01	No. 13 250 VDC Battery Operability Check (Division 1)	Revision 8
	Operations Manual:	
B.9.8	- Emergency Diesel Generators	
B.8.1.2	- EDG Emergency Service Water	
B.08.05	- Fire Protection System	
	USAR:	Revision 18
Section 8.4.1	- Safeguards EDG Systems	
Section 10.4.4	- Emergency Service Water System	

1R23 Temporary Plant Modifications

WO 0201965	Investigate/Repair/Replace Regulating Valve 6-12A M/A Station	
WO 0201982	Investigate/Repair 'A' FRV M/A Station Control Loop	
J/B 02-16	Temporary Signal Recorder for FRV 'A'	Revision 0
SCR 02-180	Temporary Signal Monitoring for FRVs	Revision 0

	Procedures and Forms:	
4AWI-04.04.03	- Bypass Control	Revision 17
4AWI-04.05.15	- Control of Troubleshooting Activities	Revision 0
C.6-CFW508	- Alarm Response - DFCS FWCV Demand Deviation	Revision 0

1EP6 Drill Evaluation

RQ-SS-38E	Loss of High Pressure Injection with Small LOCA	Revision 0
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	Revision 2
	Monticello Emergency Plan	Revision 20
5790-102-02	Monticello Emergency Notification Report Form	Revision 25
M-7-414L-007	Shift Emergency Communicator Simulator Training	Revision 2*
3389	Event Notification	Revision 40

2OS1 Access Control to Radiologically Significant Areas (71121.01)

RWP 73	Radiological Controlled Areas Excluding Locked High Radiation Areas	Revision 2
RWP 155	947 Rad Waste - Laundry Drain Tank and Pump Room	Revision 0
RWP 159	1027 Reactor - RX Building Crane	Revision 0
CR20017948	Percons-Workers Shoe Contaminated to 45,000 cpm	December 12, 2002
CR20017949	A 20 mr/hr Smear brought to Chemistry Lab	January 12, 2002
CR20018303	Postings-Investigate to Determine if Adverse Trends Exist with Radiological Postings	February 13, 2002
CR20020253	Surveys-Inadequate Post Decon Survey in MVPR	January 16, 2002
CR20021119	Percon-Rate of Low-level Personnel Contaminations in 2001 RFO was Higher than 2000 RFO	February 28, 2002
CR20021322	Question Raised about the Non-alarm of Guard House Monitor when 0.33 uCi Cs-137 Source was Carried	February 15, 2002
CR20023790	Individual did not Report a Nuclear Medical Treatment/Diagnostic Procedure	April 23, 2002

CR20024368	RAM Control-Fixed Contamination found on Pip Plug Installed in Non-Radioactive System (Fire Hose Station)	May 14, 2002
CR20024560	Individual Exited Controlled Area Portal/RCA with out Authorization From RPC	May 23, 2002
CR20025034	Postings-Worker Moved Radiological Barrier with RP Authorization	June 3, 2002
CR20025644	Proc Control-The Location Maps for Special Status Signs was not Updated in Accordance with R.07.02	June 19, 2002
CR20025652	Proc Control-Questioned By NRC Why 5 HEPA Units are Unaccounted	June 19, 2002
MNGP R.06.09	Storage and Inventory of Radioactive Material Outside the Power Block	Revision 7
MNGP R.07.02	Area Posting, Special Status Signs and Hot Spot Stickers	Revision 17
MNGP R.12.12	Vacuum Cleaner and HEPA Usages in the Radiological Controlled Area	Revision 8
	Chemistry and Radiation Protection Effectiveness Report - 1st Quarter 2002	April 12, 2002

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

CR20017583	Actual Exposure is Greater than 125% of Estimate for RWP 185	January 5, 2002
CR20017584	Actual Exposure is Greater than 125% of Estimate for RWP 177	January 5, 2002
CR20017585	Actual Exposure is Greater than 125% of Estimate for RWP 248	January 5, 2002
CR20017586	Actual Exposure is Greater than 125% of Estimate for RWP 249	January 5, 2002
CR20017645	Actual Exposure is Greater than 125% of Estimate for RWP for Drywell Leak Rate Testing	January 5, 2002
CR20017766	Actual Exposure is Greater than 125% of Estimate for RWP 537	January 5, 2002
CR20017932	Actual Exposure is Greater than 125% of Estimate for RWP 507	January 7, 2002

CR20017949	Actual Exposure is Greater than 125% of Estimate for RWP 529	January 7, 2002
CR20017968	Actual Exposure is Greater than 125% of Estimate for RWP 555	January 7, 2002
CR20017971	ALARA-Investigate to determine if There are any Trends Associated with RWPs that Exceed their Estimate by 125%	January 29, 2002
CR20017996	Actual Exposure is Greater than 125% of Estimate for RWP 226	January 7, 2002
CR20017997	Actual Exposure is Greater than 125% of Estimate for RWP 718	January 7, 2002
CR20020831	Site Dose Goal Exceeded After Person-rem Total was Adjusted for TLD Results	March 29, 2002
CR20021003	Electronic Dosimeters Under-Respond with Respect to TLDs	March 4, 2002
CR20021317	ALARA-Exposure to Engineering Personnel Higher than Expected during HPCI Run	May 14, 2002
CR20021318	ALARA-Higher than Expected Radiation Protection Personnel Exposure Received during HPCI Run	February 19, 2002
CR20024619	Accuracy of Collective Site Dose Estimates as Communicated to Site Personnel Dose Not Meet Management Expectations	May 28, 2002
CR20025186	Adverse Trend-Site Dose Trend and Projection Indicates Site Goal of 40 Rem may not be Met at Year End	June 3, 2002
CR20025697	Proc Control-ALARA Post Job Reviews not Performed on Three Jobs Meeting the Requirements of 4AWI-08.04.08	June 20, 2002
MNGP R.01.01	RWP Preparation and Issuance	Revision 33
MNPG 4AWI-08.04.08	ALARA Plan	Revision 3
NUREG-0713	Occupational Radiation Exposure at NRC Licensed Facilities - Thirty-Third Report 2000	
OR2002-002-5-035	Nuclear Oversight Observation Report - Radiological Protection	June 17, 2002

OR2002-002-5-038	Nuclear Oversight Observation Report - Radiological Protection	June 12, 2002

4OA1 Performance Indicator Verification

	Monticello Performance Indicator Data Summary Report - 1st Quarter 2002	April 22, 2002
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	Revision 2
3530-11	NRC Performance Indicator Initiating Events Worksheet (2nd Quarter 2001 - 1st Quarter 2002)	Revision 1
3530-12	NRC Performance Indicator Drywell Equipment Drain Leakage Worksheet (2nd Quarter 2001 - 1st Quarter 2002)	Revision 0
3530-7	Performance Indicator Drywell Equipment Sump Worksheet (2nd Quarter 2001 - 1st Quarter 2002)	Revision 1

4OA2 Identification and Resolution of Problems

	Design Basis Documents:	
T.5	- External Flooding	Revision 3
T.8	- Internal Flooding	Revision 2
NSPLMI-95001	Individual Plant Examination of External Events (IPEEE)	Revision 1
NSPNAD-92003	Individual Plant Examination (IPE)	Revision 0
	USAR:	Revision 18
Section 12.2.1.7.1	- External Flooding	
Section 12.2.1.7.2	- Internal Flooding	
Section 2.4.1	- Surface Water	
Section 1.3.1.4	- Hydrology	
	Operations Manual:	
A.6	- Acts of Nature	Revision 13*
B.7.1	- Liquid Radwaste	
	Procedures:	
4AWI-04.02.01	- Housekeeping	Revision 6
CR 20022403	The EDG Room Backwater Check Valves Have Not been Inspected and Cleaned Since 1995. Corrective Actions For Prior Issue Not Completed	

NRC IN 83-44, Supplement 1	Potential Damage to Redundant Safety Equipment as a Result of Backflow Through the Equipment and Floor Drain System	August 30, 1990
CR 20010976	Torque Values on the Pocket Torquing Guide Do Not Correspond to ASME Section XI Torquing Values	
CR 20013355	Pocket Torquing Guide Missing Torque Values for B16 Studs	
CR 20013746	Pocket Torquing Guide Missing Torque Values for Studs Marked 630	
CR 20020689	Job Leadership/Coordination Roles Not Adequately Defined Causing Delays and Confusion In MO-2076 Repair Work	
CR 20020647	Time Sequence From Problem Identification to In-Field Work, Regarding MO-2076 WO 0200320	
CR 20020629	MO-2076 Declared Inoperable, Dual Indication Showed on Valve When Given an Open Signal	
CR 20023642	MO-2076 Failed to Fully Open During Step 41 of Test 0062, RCIC Steam Line High Area Temperature Test and Calibration	
CR 20025312	Additional Results from Assessment O Trends in the Jumper Bypass and Modification Processes	
CR 20024362	Apparent Adverse Trend of Unauthorized Plant Modifications	
CR 20025310	Condition Report Process Not Followed Correctly; CR 20010052 Describes a Specific Action. But No ACC Was Created or Done	
CR 20025302	Jumper Bypass Requirement to Use J/B Process for Test Devices Installed >24 Hrs in Unclear	
CR 20011562	Jumper Bypass Form #3034 Not Done for Proc #8253 (Defeat DW Airlock Interlock) Which Is Required When Installed >7 Days	
CR 20012551	Test Equipment for RCIC Testing Nor Removed Within 24 Hours as Required per Bypass Control 4 AWI-04.04.03, Sect. 2.2.7	
CR 20016287	Evaluate 3rd Quarter Modification Adverse Trend. (3rd Quarter Process Performance Panel)	

CR 20016848	Jumper/Bypass Installed Despite Not Meeting Applicability Statement in Section 2.1 of 4 AWI-04.04.03	
CR 20020573	Undocumented Modification Made to the Mechanical Pressure Regulator During the 1973 Turbine Outage	
CR 20017978	PCS AM31 Input Module Power Was Disconnected as Part of Modification 01Q050	
CR 20021532	WO-0201477 Was Replacing a Cable Without a Alteration or Modification	
CR 20021599	Cables Were Replaced Without Following the AWIs for Modification or Plant Configuration or Condition Reports	
CR 20021988	Turbine Vibration Trip Operability in B.06.01-05 Was Not Properly Addressed in Bypass Eval (J/B 02-007). Vol F Issued	
CR 20021422	Multiple Site Performance Errors Associated with the Modification Process Indicates Adverse Trend	
CR 20021002	Documentation of Question by NRC Resident Regarding ESW Pump Spray Hoods and Modification Work Planned for 02/05/02	
CR 20017515	Air Bladders Utilized In MSIV Lines for Secondary Containment Without Using Jumper Bypass Process	
4 AWI-10.01.03	Condition Reporting Process	Revision 18

4OA3 Event Follow-up

CR 20020457	Reactor Scram 113 While at 100 Percent Power	
CR 20020573	Undocumented Modification Made to the MPR During the 1973 Turbine Outage	
NEI 96-01	Guidelines for 10 CFR 50.59 Evaluations	Revision 1
RG 1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	November 2000
CR 20021013	Documentation of NRC Resident Question Regarding the Application of Technical Specification Deviations in As-Found Acceptance Criteria	

CR 20022889	Received Primary Containment Group 3 Isolation Signal on HI Flow When Repressurizing RWCU Piping	
B.01.04	Operations Manual - Reactor Recirculation System	
C.4.F	Rapid Power Reduction	Revision 13

4OA7 Licensee Identified Violation

CR 20024417	Door 46 Found Open With No Personnel in Attendance. This Is a Repeat of CR 20020510	
CR 20021047	Door 45 and Door 46 Have Multiple Signs Attached, Including "Caution" Signs With Instructions	
CR 20020510	Reactor Building Railroad Door 46 Found Open When No Work Was In Progress. This Is Not Allowed per the Caution Sign on the Door	
CR 20016321	Reactor Building Railroad Doors (Inner) Were Found Open With No Work in Progress and Unattended, Contrary to Posted Signs	
CR 19961165	Reactor Building Inner Railroad Door Control	
AG 1996-O-2	QA Internal Audit Report on Plant Operations	August 23, 1996