March 30, 2006

Mr. J. Conway
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 LICENSE RENEWAL SCOPING, SCREENING, AND AGING MANAGEMENT INSPECTION REPORT 05000263/2006006(DRS)

Dear Mr. Conway:

On February 22, 2006, the NRC completed an inspection regarding your application for license renewal for the Monticello Nuclear Generating Plant. The enclosed report documents the inspection results, which were discussed on February 22, 2006, with members of your staff in an exit meeting open for public observation at the Monticello Community Center in Monticello, Minnesota.

The purpose of this inspection was an examination of activities that support the application for a renewed license for Monticello. The inspection addressed the processes of scoping and screening plant equipment to select equipment subject to an aging management review and development and implementation of aging management programs to support a period of extended operation. As part of the inspection, the NRC examined procedures and representative records, interviewed personnel, and visually examined accessible portions of various systems, structures or components to verify license renewal boundaries and to observe any effects of equipment aging.

The inspection concluded that the scoping, screening, and aging management license renewal activities were generally conducted as described in the license renewal application, as supplemented through your responses to requests for additional information from the NRC. The inspection also concluded that documentation supporting the application is generally in an auditable and retrievable form. Existing aging management programs were determined to be functioning adequately and, when all the programs are implemented as described in your license renewal application, there is reasonable assurance that the intended functions of vital plant systems, structures, and components will be maintained through the period of extended operation.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and any response you provide will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-263
License Nos. DPR-22

Enclosure: Inspection Report 05000263/2006006(DRS)
w/Attachment: Supplemental Information

cc w/encl: M. Sellman, Chief Executive Officer
and Chief Nuclear Officer
Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
Commissioner, Minnesota Department of Health
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
Manager - Environmental Protection Division
Minnesota Attorney General's Office

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-263
License Nos:	DPR-22
Report No:	05000263/2006006
Applicant:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello Nuclear Generating Plant Nuclear Management Company, LLC 2807 West County Road 75 Monticello, MN 55362-9637
Dates:	January 23 through February 22, 2006
Inspectors:	P. Lougheed, Senior Engineering Inspector (lead) Z. Falevits, Senior Engineering Inspector M. Holmberg, Senior Engineering Inspector J. Neurauter, Senior Engineering Inspector D. Merzke, License Renewal Project Manager, NRR C. Acosta-Acevado, Engineering Inspector
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000263/2006006(DRS); Nuclear Management Company; on 01/23/2006 - 02/22/2006; Monticello Nuclear Generating Plant; License Renewal Inspection Program, Scoping, Screening, and Aging Management Programs

This inspection of the applicant's license renewal aging management review was performed by five regional office inspectors and one staff member from the NRC's office of Nuclear Reactor Regulation. The team used NRC Manual Chapter 2516 and NRC Inspection Procedure 71002 as guidance for performing this inspection. No “findings” as defined in NRC Manual Chapter 0612 were identified.

The team concluded that, in general, the applicant performed their license renewal scoping, screening, and aging management review in accordance with the Monticello license renewal application. No impediments to the applicant receiving an extended operating license were identified.

REPORT DETAILS

A. Inspection Scope

This inspection was conducted by NRC Region III inspectors and the license renewal project manager from the Office of Nuclear Reactor Regulation (NRR). The inspection was performed in accordance with NRC Manual Chapter 2516 and NRC Inspection Procedure 71002, “License Renewal Inspection,” dated February 18, 2005.

This inspection looked at both the applicant’s scoping and screening methodology and aging management programs, as described in the license renewal application (LRA), submitted to the NRC on March 24, 2005.

The attachments to this report list the applicant personnel contacted, the documents reviewed, and the acronyms used.

B. Visual Observation of Plant Equipment

During this inspection, the inspectors performed walkdown inspections of portions of many of the plant systems, structures, and components (SSCs), including some SSCs which were outside the scope of license renewal (LR). The walkdowns were intended to determine the acceptability of the scoping boundaries, to observe the current condition of the SSCs, and to assess the likelihood that a proposed aging management program would successfully manage any aging effects. Specific comments on the walkdown results are presented in the sections below. Portions of the following systems were walked down:

- Condensate and Feedwater Systems;
- Diesel Generator Starting Air System;
- Electrical Power Distribution System;
- Emergency Power Distribution System;
- Fire Protection System;
- Fuel Pool Cooling System;
- Heating and Ventilation System;
- High Pressure Coolant Injection System;
- Instrument and Service Air Systems;
- Liquid and Solid Radwaste Systems;
- Main Condenser;
- Non-Essential Diesel;
- Reactor Building Closed Cooling Water System;
- Reactor Water Cleanup System; and
- Service and Seal Water Systems.

The following structures were walked down:

- Auxiliary Building;
- Block Walls;
- Buried Cable Manholes;
- Electrical Penetrations;
- Emergency Diesel Generator Rooms;
- Exterior Equipment Foundations;
- High Pressure Coolant Injection Pump Room;
- Intake Structure Interior Area;
- Overhead Cranes;
- Non-Segregated Bus Duct;
- Pipe Supports;
- Plant Power, Control and Instrumentation Cable Installations;
- Reactor Core Isolation Cooling Pump Room;
- Residual Heat Removal Pump Room;
- Torus Room Area;
- Turbine Building;
- Substation;
- 4.16 kV Switchgear Rooms; and
- 985 Pump Room.

The inspectors reviewed videotaped examinations of the following reactor vessel internals conducted by the applicant during the 2003 and 2005 refueling outages:

- Core Shroud Welds;
- Core Spray Headers and Spargers;
- Core Spray Piping “T-Box” Repair Hardware;
- Core Spray Piping Welds;
- Jet Pump Components;
- Shroud Support Welds;
- Steam Dryer Support Lug Attachment Welds;
- Steam Dryer Welds; and
- Surveillance Sample Holder Attachment Welds.

C. <u>Review of Scoping and Screening Methodology</u>

In order to assess the applicant’s scoping and screening methodology, the inspection concentrated on those non-safety-related systems whose failure could prevent safety-related SSCs from accomplishing a safety function, in accordance with 10 CFR 54.4(a)(2). The inspection specifically focused on the SSCs, or portions of SSCs, which the applicant had determined to be outside the scope of license renewal. To verify that non-safety-related SSCs were correctly captured within or omitted from the scope of license renewal, the inspectors reviewed LR documents, interviewed personnel, and walked down the selected SSCs.

1. Condensate and Feedwater

The condensate and feedwater (CFW) system is primarily a non-safety-related system which supplies condensate from the main condenser to the reactor vessel at an elevated temperature and pressure. The CFW system includes the condensate demineralizer, the reactor feedwater pump seal, and the zinc injection passivation subsystems. Some components in the CFW system are within the scope of LR because they perform a safety-related function, in accordance with 10 CFR 54.4(a)(1). Some components are within scope because their failure could affect the capability of safety-related components to perform their safety function, in accordance with 10 CFR 54.4(a)(2). In addition, some components are within scope because they support station blackout, in accordance with 10 CFR 54.4(a)(3).

The inspectors reviewed the LR boundary drawings, the application, and the updated safety analysis report (USAR), and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the condensate and feedwater systems. The inspectors concluded that the applicant had performed scoping and screening for the condensate and feedwater systems in accordance with the methodology described in the LRA and the rule.

2. Diesel Generator Starting Air

The purpose of the diesel starting air system is to provide the motive force to initially put the diesel engine in motion and begin the diesel cycle. The diesel engine is started by compressed air stored in the starting air system receiver tanks. In-scope components are located between the air compressor discharge check valves and the diesel engine air start motors. The air compressors and driers are not in-scope of LR because they are not required to start the diesel engine.

The inspectors reviewed the LR boundary drawings, the application, the scoping and screening reports and the USAR, and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the diesel starting air system. The inspectors concluded that the applicant had performed scoping and screening for the diesel starting air system in accordance with the methodology described in the LRA and the rule.

3. Drywell Atmospheric Cooling

The drywell atmospheric cooling system is a non-safety-related system which the applicant considered outside the scope of license renewal. The system provides normal cooling to the drywell to maintain the bulk average drywell ambient temperature less than 135EF and localized temperatures below 150EF during normal plant operation. This assures a sustained life for insulation and gasket materials or sealants inside the drywell. The system does not have any safety function and is not required for any regulated event.

The inspectors reviewed the USAR, the scoping and screening output report, and the operations manual for the drywell atmospheric system. The inspectors also reviewed drawings of the system and held discussions with applicant personnel. The inspectors agreed with the applicant's assessment that this non-safety-related system did not meet the requirements to be included within the scope of license renewal.

4. Electric Power Distribution System

The electric power distribution system provides alternating-current power to both safety-related and non-safety-related SSCs in the plant. Components were placed into scope either because they individually had a safety-related function or were required to perform during a regulated event. Additionally, some classes of components were placed in scope on a commodities basis. The majority of the electrical components screened out as active and did not require an aging management program.

The inspectors reviewed the related electrical LR boundary drawing, the application, and the applicable USAR sections and interviewed personnel responsible for the electrical power distribution system and the LR program. The inspectors also performed system walkdowns of accessible and partially inaccessible portions of the electrical power distribution system.

The inspectors identified discrepancies between the scoping and screening report and electrical boundary drawing number LR-36298. Specifically, the LR classification of breakers fed from several 480 Volt load centers were not consistent with the LR scoping and screening document. The applicant also identified several other load center cubicles which were not previously shown as being in-scope, but which should have been. The applicant determined that the additional components brought into scope of LR were all active components and therefore screened out and did not required aging management. Therefore, there were no new aging management concerns.

The inspectors concluded that the applicant had performed scoping and screening for the electrical power distribution system in accordance with the methodology described in the LRA and the rule.

4. Fuel Pool Cooling and Cleanup

The purpose of the fuel pool cooling and cleanup system is to remove decay heat generated and to maintain pool water purity and clarity. Components in the fuel pool cooling and cleanup system are non-safety-related and their failure could affect the capability of safety-related SSCs to perform their safety function; therefore, they are in-scope in accordance with 10 CFR 54.4(a)(2).

The inspectors reviewed the LR boundary drawings, the application, and the USAR, and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the fuel pool cooling and cleanup system. The inspectors concluded that the applicant had performed scoping and screening for the fuel pool cooling and cleanup system in accordance with the methodology described in the LRA and the rule.

5. Heating and Ventilation

The purpose of the heating and ventilation (HTV) system is to provide appropriate ambient environmental conditions for plant safety-related equipment, specifically for the high pressure coolant injection and core spray systems. Another purpose is to provide for controlled flow direction and release of radioactive gases during non-accident conditions. In addition, components of the system are included in the secondary containment isolation function. There are some portions of the HTV system which perform a safety-related function, in accordance with 10 CFR 54.4(a)(1). The HTV system also is credited for use in mitigating the regulated events of fire protection and environmental qualification (EQ).

The inspectors reviewed the LR boundary drawings, the application, and the USAR, and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the heating and ventilation system. As a result of the system walkdowns, the inspectors identified additional piping and valves which should have been included within the scope of LR in accordance with 10 CFR 54.4(a)(2). One portion of piping along with two valves and a steam trap inside the emergency diesel generator room, as well as a section of non-safety-related piping attached to safety-related piping outside the torus were brought into scope. The inspectors concluded that the applicant had performed scoping and screening for the HTV system in accordance with the methodology described in the LRA and the rule.

6. Instrument Air

The instrument air system is a non-safety-related system which provides the plant with a continuous supply of oil-free compressed air. The instrument air portion of the system supplies dried compressed air for most of the pneumatic instruments and controls in the plant. The instrument air system is not required post-accident since equipment requiring compressed air for operation during or immediately subsequent to an accident receives air from local accumulators or other pneumatic sources. However, the instrument air system interfaces with safety related systems resulting in some structures and components within this system being in-scope for license renewal. Components in the instrument air system whose failure could affect the capability of safety related components to perform their safety function, are in-scope for license renewal. In addition, some structures and components are in-scope due to environmental qualification. The remainder of the system was excluded because it was considered not to perform a safety-related function, not to potentially impact the function of another safety system, and not provide a function related to one of the regulated events.

The inspectors reviewed the LR boundary drawings, the application, the scoping and screening reports and the USAR, and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the instrument and service air system. The inspectors identified that LR boundary drawings LR-36049-10 at locations D-6 and B-6 showed the lines continuing to drawing LR-36049-12 at location B-6. However, the continuation line on drawing LR-36049-12 at location B-6 LR did not show the lines as being in-scope. The inspectors determined through walkdowns and discussion with the applicant that the lines should be in-scope until a physical boundary was reached. Based on this clarification, the inspectors concluded that the applicant had performed scoping and screening for the instrument and service air system in accordance with the methodology described in the LRA and the rule.

7. Liquid and Solid Radwaste

The liquid and solid radwaste systems are non-safety-related systems designed to collect, process, and dispose of radioactive and potentially radioactive wastes in a controlled and safe manner without limiting plant power output or availability. The liquid radwaste system is designed to accommodate the radioactive input resulting from the design basis maximum fuel leakage condition. The solid radwaste system is also designed to package, store, monitor, and provide shielded storage facilities to allow for radioactive decay and for temporary storage prior to shipment for off-site disposal. Some components in these systems were in-scope in accordance with 10 CFR 54.4(a)(1) and other components were brought into scope due to supporting the environmental qualification regulated event in accordance with 10 CFR 54.4(a)(3). Additionally, since the failure of some non-safety-related components could affect the capability of safety-related components to perform their safety function, those components were brought into scope in accordance with 10 CFR 54.4(a)(2).

The inspectors reviewed the LR boundary drawings, the application, the scoping and screening output report, and the USAR. The inspectors also performed system walkdowns of accessible portions of the liquid and solid radwaste systems, with an emphasis on the boundaries where the applicant determined the systems would no longer be in-scope for license renewal. Additionally, the inspectors interviewed personnel responsible for the systems and familiar with the LR process. The inspectors determined that, in some cases, the LR boundary drawings did not correctly show the break point where radwaste SSCs transitioned from being in or out of scope. The applicant noted each of these items and wrote license renewal action items to clarify the boundary location.

Additionally, the inspectors noted that the application did not correctly describe the basis for concluding that portions of the liquid radwaste system inside the reactor building were out of scope for license renewal. Specifically, the application stated that all radwaste system components existing in either the turbine or reactor buildings, and constituting a liquid pressure boundary, were in-scope. However, the inspectors noted that the portions of the liquid radwaste system inside a room – designated as the 985 pump room – were considered out of scope in the scoping and screening output report and on the LR boundary drawings. The inspectors determined that the 985 pump room was physically inside the reactor building. Furthermore the room opened to and could only be accessed from the reactor building. Therefore, the inspectors questioned the dichotomy between the statement in the application and the boundaries described in the other documents.

The inspectors performed a physical walkdown of the 985 pump room and its vicinity. The inspectors acknowledged that the 985 pump room was reasonably physically separate from the rest of the reactor building, that all systems and components inside the 985 pump room were non-safety-related and that failure of the components within the room would not adversely impact any safety-related components. Therefore, the inspectors agreed that components inside the 985 pump room did not need to be in-scope for LR under the provisions of 10 CFR 54(a)(2). The applicant stated that a change would be made to the application wording to explain that components in the 985 pump room were out of scope for license renewal. The applicant planned to include the revision in the annual update to the application. With this revision, the inspectors concluded that the applicant had performed scoping and screening for the liquid and solid radwaste systems in accordance with the methodology described in the LRA and the rule.

8. <u>Main Condenser</u>

The purposes of the main condenser are to provide a heat sink for the steam cycle, to remove non-condensable gases, and to serve as a central collection point for system drains. The system is non-safety-related, but is credited for post-accident plate out and holdup of radioactive iodine in the loss of coolant accident and control rod drop accident analyses in the USAR. Therefore, the applicant placed portions of the system in scope for license renewal.

The inspectors reviewed the LR boundary drawings, the application, and the USAR, and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the main condenser system. The inspectors concluded that the applicant had performed scoping and screening for the main condenser system in accordance with the methodology described in the LRA and the rule.

9. <u>Non-Essential Diesel</u>

The non-essential diesel is a non-safety-related component which the applicant considered outside the scope of license renewal. The diesel provides a source of power for the safety parameter display system, should normal power be lost. Additionally, plant procedures allow the operators to use the diesel during a station blackout. However, the non-essential diesel is not credited in the station blackout analysis and does not perform any safety-related function.

The inspectors reviewed the USAR, the scoping and screening output report, the operations manual, and abnormal procedures for the non-essential diesel. Additionally, the station blackout analysis design basis document was reviewed and a walkdown of the area was performed. The inspectors agreed with the applicant’s assessment that this non-safety-related system did not meet the requirements to be included within the scope of license renewal.

10. Reactor Water Cleanup

The reactor water cleanup system is a filtering and ion exchange system that maintains water purity in the reactor and recirculation lines during all modes of plant operation. This non-safety-related system provides for continuous purification of a portion of the reactor recirculation system flow with a minimum of heat loss and water loss from the cycle. Portions of the system perform a safety function and are within the scope of LR in accordance with 10 CFR 54(a)(1). Other components were identified as being required for the regulated events of anticipated transient without scram and environment qualification and so were in-scope under 10 CFR 54(a)(3). Finally the failure of some non-safety-related components could affect the function of safety-related components, putting some components in-scope under 10 CFR 54(a)(2).

The inspectors reviewed the LR boundary drawings, the application, and the scoping and screening output report. The inspectors also performed system walkdowns of accessible portions of the reactor water cleanup system, with an emphasis on the boundaries where the applicant determined the system would no longer be in-scope for license renewal. Additionally, the inspectors interviewed personnel responsible for the systems and familiar with the LR process. The inspectors determined that portions of the reactor water cleanup system physically located within the 985 pump room were shown in scope. Upon further review, the applicant confirmed that those portions were non-safety-related and did not affect the classification of the 985 pump room and components contained therein as being out-of-scope for license renewal. Similar to the description for the radwaste systems, the application did not describe the components within the 985 pump room as being out-of-scope. The application revision described above would correct this issue as well. Therefore, the inspectors concluded that the applicant had performed scoping and screening for the reactor water cleanup system in accordance with the methodology described in the LRA and the rule.

11. Reactor Vessel Internals

The reactor pressure vessel internals consists of all the structures and components within the reactor vessel that provide support for the core, control rod system support, instrumentation support, steam quality enhancement, and that direct coolant flow. The portions of the reactor pressure vessel internals containing components subject LR include the core shroud and core plate, top guide, core spray lines and spargers, jet pump assemblies, fuel support and control rod drive housing, and guide tubes. The nuclear fuel is not addressed because the fuel is periodically replaced thereby making it short-lived. Other core internal components which were not considered within scope of 10 CFR 54.4(a)(1)(2) or (3) included the steam separator assembly, the feedwater spargers, and the surveillance sample holders.

The inspectors reviewed the LRA and the USAR, and boiling water reactor (BWR) vessel internals project (VIP) document BWRVIP-06 “BWR Vessel Internals Project Safety Assessment of BWR Internals.” Based upon these reviews, the inspectors did not identify any discrepancies in the applicants scoping and screening for the reactor vessel internals. Therefore, the inspectors concluded that the applicant had identified the appropriate core internals components subject to aging management in accordance with the methodology described in the LRA and the rule.

12. Service and Seal Water

The purpose of the service and seal water (SSW) system is to supply screened and strained raw cooling water from the Mississippi river to various non-essential plant heat loads and services during all modes of operation. The seal water portion of the SSW system provides filtered well water to the shaft seals for various pumps including the service water pumps, residual heat removal service water (RSW) pumps, and the circulating water pumps. The service water portion is used to keep the RSW subsystem filled and pressurized during normal plant operation, and serves as a backup supply for RSW motor thrust bearing oil coolers. The SSW system also supplies water to the sodium hypochlorite subsystem. The SSW system is normally in service during plant operation and shutdown. The SSW system is not required during or immediately subsequent to a design basis accident and is not safety related.

The SSW system lines have the potential for spatial interactions with safety related equipment. Also, portions of service water piping were upgraded to Class I seismic requirements for internal flooding concerns. The SSW system is connected to the fire water system and supplies water to the fire system jockey pump. SSW system valves provide a pressure boundary to prevent backflow from the fire system; therefore these portions of the SSW system were placed in scope for the regulated event of fire protection. The remainder of the system was excluded because it was considered not to perform a safety-related function, not to potentially impact the function of another safety system, and not provide a function related to one of the regulated events.

The inspectors reviewed the LR boundary drawings, the application, the scoping and screening reports, and the USAR, and interviewed personnel responsible for the program. The inspectors also performed system walkdowns of accessible portions of the SSW system. During review of the LR boundary drawing LR-36665, the inspectors noticed that a continuation of in-scope non-safety-related SSW piping at location C-5 occurred going to location A-5 on the same drawing. However, the continuation at location A-5 was not identified as within scope of license renewal. Following discussion and a plant walkdown, the inspectors determined that the continuation line needed to be shown as in-scope for LR up to the first physical barrier. With this minor issue corrected, the inspectors concluded that the applicant had performed scoping and screening for the SSW system in accordance with the methodology described in the LRA and the rule.

D. Review of Aging Management Programs

The inspection assessed the adequacy of current implementation of existing aging management programs (AMPs) credited in the applicant's LR program. This included verification that current AMPs would ensure that aging effects would be managed so that there was reasonable assurance that an SSCs intended function would be maintained throughout the period of extended operation. For those programs indicated by the applicant as being consistent with NUREG 1801, “Generic Aging Lessons Learned (GALL) Report,” the inspectors confirmed that the applicant’s program included the GALL attributes. For those programs which the applicant indicated were new or being enhanced, the inspectors confirmed that commitments existed and were sufficient to support future implementation. For those programs where the applicant indicated that they intended to take exception to the GALL, the inspectors reviewed the exceptions against the GALL recommendations and evaluated the acceptability of the applicant’s proposal.

The inspection also consisted of walkdowns of selected in-scope SSCs to assess how plant equipment was being maintained under the current operating license and to visually observe examples of non-safety-related equipment determined to be in scope due to their proximity to safety-related equipment and their potential for failure due to aging effects.

1. <u>10 CFR Part 50, Appendix J (B2.1.01)</u>

The 10 CFR Part 50, Appendix J program is an existing program that is consistent with the recommendations of NUREG-1801, Section XI.S4, “10 CFR Part 50, Appendix J.” However, the applicant also identified some exceptions to the GALL program. The 10 CFR Part 50, Appendix J program specifies pneumatic pressure tests and visual examinations to verify the structural and leak tight integrity of the primary containment.

An overall (Type A) pressure test assesses the capacity of the containment to retain design basis accident pressure. This test also measures total leakage through the containment pressure-retaining boundary. Local (Type B and C) tests measure leakage through individual penetration isolation barriers. These barriers are maintained as required to keep overall and local leakage under Technical Specification and plant administrative limits.

The inspectors reviewed the LR program basis documentation, aging management review documents, existing plant procedures, and recently completed inspection results. The inspectors reviewed the applicant’s plant-specific operating experience through a corrective action program search for degraded penetrations which exceeded the administrative leakage limits and verified that the applicant performed adequate historic reviews of plant-specific experience to determine aging effects.

The inspectors concluded that the 10 CFR Part 50, Appendix J program effectively manages aging effects. Continued implementation of the 10 CFR Part 50, Appendix J program will provide reasonable assurance that the aging effects will be managed so that the structural components within the scope of the program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

2. <u>ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.02)</u>

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (the ASME Code) Section XI inservice inspection (ISI), subsections IWB, IWC, and IWD program is an existing program that is generally consistent with NUREG 1801, Section XI.M.1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD". However a number of exceptions were identified during or following the NRC aging management review audit. This program provides for condition monitoring of Class 1, 2, and 3 pressure retaining components and their integral attachments.

The inspectors reviewed LR program basis documents, program health report, inservice inspection self-assessment report, nondestructive examination records focused on Section XI Code-required reactor vessel nozzle examinations, NRC inspection report findings, and licensee event reports (LERs) associated with pressure retaining Code components. The inspectors also searched the applicant's corrective action program records for degraded Code components to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP. Additionally, the inspectors did not identify any additional exceptions from the Section X1.M.1 program. Therefore, the inspectors concluded that the applicant's AMP should continue to assure the ASME Code pressure boundary function consistent with the current licensing basis for the period of extended operation.

The inspectors identified a current plant issue associated with the applicant's failure to submit relief requests to the NRC for six reactor vessel nozzle examinations with limited weld volumetric coverage completed during the 2000 refueling outage. The applicant captured this issue in AR 01013875 and the inspectors turned this issue over to the NRC resident inspector for followup.

3. <u>ASME Section XI, ISI Subsection IWF (B2.1.03)</u>

The ASME Code Section XI ISI, Subsection IWF program is an existing program that is consistent with NUREG 1801, Section XI.S.3, "ASME Section XI Inservice Inspection, Subsections IWF" with one exception and with enhancements as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). The program provides for condition monitoring of Class 1, 2, and 3 and MC component supports.

The inspectors reviewed LR program basis documents, NRC inspection report findings, and LERs associated with ASME Code Class 1, 2, and 3 and MC component supports. The inspectors also searched the applicant's corrective action program records for degraded Code component supports to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors identified that the existing program procedures lacked specific guidance to confirm the acceptability of baseplate gaps (e.g., gap between the metal support base plate and the concrete support structure) for Code component supports and that no exception was taken for these gaps. The applicant wrote action requests (ARs) 00829856-02 and 00830329-02 to incorporate procedural guidance for support plate baseplate gaps. The inspectors concluded that when these requirements were incorporated into existing procedures, the applicant's AMP should have adequately managed current plant aging effects relating to Code Class 1, 2, and 3 and MC support integrity consistent with the current licensing basis for the period of extended operation.

4. <u>Bolting Integrity (B2.1.04)</u>

The bolting integrity program is an existing program which complies with the recommendations of NUREG-1801, Section XI.M18, "Bolting Integrity," with some exceptions. The program manages the aging affects associated with bolting in the scope of LR through periodic inspection, material selection, thread lubricant control, assembly and torque requirements, and repair and replacement requirements. These activities are based on the applicable requirements of ASME Section XI and plant operating experience and includes consideration of the guidance contained in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," EPRI documents NP-5769, "Degradation and Failure of Bolting in Nuclear power Plants", TR-104213, "Bolted Joint Maintenance and Application Guide," and NP-5067, Volumes 1 and 2, "Good Bolting Practices." The program also credits other aging management programs for inspection of installed bolts; some of these programs require enhancement to include such inspections.

The inspectors reviewed LR program basis documentation, aging management review documents, and existing plant procedures. The inspectors witnessed a bolt torquing activity and verified that the aging management attributes were employed. Additionally, the inspectors performed a detailed walkdown of the high pressure coolant injection system and the torus exterior to verify the adequacy of structural bolting. Finally, the inspectors performed numerous searches of the corrective action program to determine the acceptability of the applicant's current program. The inspectors concluded that the bolting integrity program effectively manages aging effects. Continued implementation of the bolting integrity program will provide reasonable assurance that the aging effects will be managed so that bolted components and structures within the scope of the program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

5. <u>Buried Piping and Tanks Inspection (B2.1.05)</u>

The buried piping and tanks inspection program is an existing program that, when enhanced, will be comparable to NUREG-1801, Section XI.M34, "Buried Piping and Tanks Inspection." The program manages the aging effects for buried piping, conduit, and tanks in-scope for license renewal. Preventive measures consist of preventive coatings and/or wraps on buried components. Condition monitoring consists of periodic inspections of buried components.

The inspectors reviewed the applicable LR program basis documentation, existing inspection procedures, and confirmed that the applicant had commitments in place to enhance the program prior to the start of the period of extended operation. The inspectors also interviewed the buried piping and tanks inspection program owner, reviewed a diesel fuel oil storage tank inspection report, reviewed an underground piping inspection report, and reviewed applicant condition reports and operating experience to verify concerns related to buried piping and tanks are being addressed through the applicant's corrective action program.

The inspectors noted that although the program basis documents indicated that buried conduit would be managed by the buried piping and tanks inspection program, the existing inspections and related procedures were limited to buried piping and the diesel fuel oil storage tank. Since buried conduit is galvanized and not wrapped or coated similar to carbon steel piping or tanks, conduit aging could be different than that for the underground piping and tanks. The applicant committed to identify buried conduit as an enhancement to the scope of the buried piping and tanks inspection program to be included in the LRA annual update.

The inspectors concluded that the buried piping and tanks inspection program, when enhanced as described in the application and with the above stated changes, will adequately manage aging effects. Implementation of the buried piping and tanks inspection program will provide reasonable assurance that the aging effects will be managed so that in-scope buried components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

6. <u>Bus Duct Inspection Program (B2.1.06)</u>

The bus duct inspection program will be a new program that, when implemented, will be consistent with the ten elements described in Appendix A of NUREG-1800, "Standard Review Plan for Review of LR Applications for Nuclear Power Plants." This AMP will demonstrate, for in scope non-segregated bus duct, that the aging effects caused by ingress of moisture or contaminants (dust debris), insulation degradation and bolt relaxation will be adequately managed to provide reasonable assurance that the non-segregated bus ducts will perform their intended function, consistent with the current licensing basis, during the period of extended operation.

The inspectors reviewed aging management program related documentation, condition reports, preventive maintenance (PM), and LR procedures and activities, and confirmed that the applicant had NRC commitment number M05019A in place to implement the program prior to the start of the period of extended operation. The inspectors also interviewed applicant engineers concerning the bus duct inspection program to determine how and when aging management program changes that are required to satisfy LR commitments and applicable interim staff guidance (ISGs) will be developed and implemented.

In addition, the inspectors conducted field inspections of accessible portions of the non-segregated bus duct and identified a number of anomalies concerning performance of past bus duct PM inspections and several material condition items. The applicant initiated AR 01013874, AR 01013360, and generic AR 008298888-02 to address the concerns noted. The applicant also planned to revise the PM procedures as part of the program development. The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects.

The inspectors concluded that the bus duct inspection program, when implemented as described, will effectively manage aging effects, since it will incorporate proven monitoring and testing techniques, acceptance criteria, corrective actions, and administrative controls. Implementation of this program will provide reasonable assurance that the effects of aging will be managed such that components within the scope of the program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

7. <u>BWR Control Rod Drive Return Line Nozzle (B2.1.07)</u>

The BWR control rod drive return line nozzle (CRDRL) program is an existing program that is consistent with NUREG 1801, Section XI.M.6, "BWR Control Rod Drive Return Line Nozzle" with two exceptions. The program provides for condition monitoring of the BWR CRDRL nozzles.

The inspectors reviewed LR program basis documents and LERs associated with the CRDRL nozzles. The inspectors also searched the applicant's corrective action program records for issues related to CRDRL nozzle integrity to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP. The inspectors also did not identify any additional exceptions from the Section X1.M.6 program. Therefore, the applicant's AMP should continue to perform its intended function and maintain the integrity of the CRDRL nozzles consistent with the current licensing basis for the period of extended operation.

8. <u>BWR Feedwater Nozzle (B2.1.08)</u>

The BWR feedwater nozzle program is an existing program that is consistent with NUREG 1801, Section XI.M.5, "BWR Feedwater Nozzle" with one exception and enhancement as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). The applicant's program provides for condition monitoring of the BWR feedwater nozzles.

The inspectors reviewed LR program basis documents and LERs associated with feedwater nozzles. The inspectors also searched the applicant's corrective action program records for issues related to feedwater nozzle integrity to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP. The inspectors also did not identify any additional exceptions from the Section X1.M.5 program. Therefore, the applicant's AMP should continue to perform its intended function and maintain the integrity of the feedwater nozzles consistent with the current licensing basis for the period of extended operation.

9. <u>BWR Penetrations (B2.1.09)</u>

The BWR Penetrations Program is an existing program that is consistent with NUREG 1801, Section XI.M.8, "BWR Penetrations" with three exceptions as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). This program provides for condition monitoring of the reactor vessel penetrations.

The inspectors reviewed LR program basis documents and LERs associated with BWR reactor vessel penetrations. The inspectors also searched the applicant's corrective action program records for issues related to vessel penetration integrity to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP nor did the inspectors identify any additional exceptions from the Section X1.M.8 program. Therefore, the applicant's AMP should continue to perform its intended function and maintain the integrity of the feedwater nozzles consistent with the current licensing basis for the period of extended operation.

10. <u>BWR Stress Corrosion Cracking (B2.1.10)</u>

The BWR penetrations program is an existing program that is consistent with NUREG 1801, Section XI.M.7, "BWR Stress Corrosion Cracking" with two exceptions as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). This program provides for condition monitoring of pressure boundary material susceptible to stress corrosion cracking.

The inspectors reviewed LR program basis documents and LERs associated with pressure boundary components affected by stress corrosion cracking. The inspectors also searched the applicant's corrective action program records for issues related to stress corrosion cracking to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP. The inspectors also did not identify any additional exceptions from the Section X1.M.7 program.

Therefore, the applicant's AMP should continue to perform its intended function and maintain the integrity of components susceptible to stress corrosion cracking consistent with the current licensing basis for the period of extended operation.

11. <u>BWR Vessel Inner Diameter Attachment Welds (B2.1.11)</u>

The BWR inner diameter (ID) attachment welds program is an existing program that is consistent with NUREG 1801, Section XI.M.4, "BWR ID Attachment Welds" with two exceptions as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). The program provides for condition monitoring of the reactor vessel interior attachment welds within and beyond the beltline region.

The inspectors reviewed LR program basis documents and videotaped visual examinations of reactor vessel ID attachment welds. The inspectors also searched the applicant's corrective action program records for issues related to vessel ID attachment welds to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP. The inspectors also did not identify any additional exceptions from the Section X1.M.4 program. Therefore, the applicant's AMP should continue to perform its intended function and maintain the integrity of the vessel ID attachment welds consistent with the current licensing basis for the period of extended operation.

12. <u>BWR Vessel Internals (B2.1.12)</u>

The BWR vessel internals program is an existing program that is consistent with NUREG 1801, Section XI.M.9, "BWR Vessel Internals" with one exception and one enhancement as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). The program provides for condition monitoring of the reactor vessel internals components for crack initiation and growth.

The inspectors reviewed LR program basis documents, videotaped visual examinations of reactor vessel internals components and LERs associated with vessel internal components. The inspectors also searched the applicant's corrective action program records for issues related to vessel internal components to determine plant specific aging effects and to assess the program's effectiveness at detecting and monitoring for age related degradation. Based upon these reviews, the inspectors identified that additional changes were necessary to ensure that the applicant adequately managed plant aging effects relating to reactor vessel internals in accordance with the AMP and the GALL.

a) The inspector noted that in-core monitoring instrument dry tubes were within the scope of the applicant's vessels internals AMP. However, these tubes were not subject to periodic inspections under the applicant's AMP which credited the BWRVIP-130 BWR water chemistry guidelines and the ASME Section XI inspection programs. Because these tubes are subject to radiation induced damage above the threshold for irradiation assisted stress corrosion cracking (IASCC), they could crack and cause pressure boundary leakage. GALL Section X1.M.9 identified that cracking has been observed at other BWRs. Furthermore, General Electric (GE) service information letter, SIL 409, "Incore Dry Tube Cracks," recommended periodic (every other outage) visual examinations focused on the upper two feet of the tube to detect cracking. The applicant had voluntarily implemented these examinations for the current license; however, the applicant had not committed to continue with these examinations during the period of extended operation within their AMP. The applicant stated that the GE SIL 409 recommended examinations of in-core monitoring instrument dry tubes would be incorporated into their AMP during the next annual LRA revision.

b) The steam dryer was within the scope of the applicant's vessels internals AMP for its structural function and the applicant conducted periodic inspection of steam dryer welds potentially subject to cracking. The applicant submitted the LRA prior to the issuance of the BWRVIP inspection program guidance defined by BWRVIP-139 "Steam Dryer Inspection and Flaw Evaluation Guidelines." The applicant stated that the BWRVIP-139 steam dryer weld examinations would be incorporated into the AMP during the next annual LRA revision.

c) In CAP 014359 (condition report 20000209), the applicant documented that during the 2000 refueling outage, areas of the steam dryer in close proximity to the main steam nozzles appeared polished and that this wear could be caused by steam impingement. The applicant also documented in AR 000032 that Vermont Yankee, with an identical steam dryer design, had observed evidence of steam erosion at the underside of the steam dryer. To evaluate if steam erosion was occurring, a degradation mechanism not identified for the steam dryer in GALL Section XI.M9, the applicant stated that the affected areas of the Monticello steam dryer would be reinspected during the next refueling outage. The inspectors also forwarded this issue to the NRR technical staff for further evaluation.

d) In two letters dated May 30, 1997, and October 30, 1997, from the chairman of the BWRVIP to the NRC, the BWRVIP committed member utilities (including Monticello) to implementation of the BWRVIP guidelines (applicable to vessel internals) to the extent possible. The applicant stated that a direct commitment to the NRC for implementation of the BWRVIP internals guidelines (similar to this existing third party commitment) would be included during the next annual LRA revision.

e) In LRA Table B1.6-11 "Responses to BWRVIP-74-A for the MNGP [Monticello Nuclear Generating Plant], Table 4-1", the applicant stated that the internal core spray piping welds P1, 2, and 3 were not inspected in accordance with BWRVIP-18 "Core Spray Internals Inspection and Flaw Evaluation Guidelines" because mechanical clamps were installed to insure the structural integrity of the sparger T-box welds and that a visual inspection was conducted each outage to confirm that T-box integrity was maintained. Specifically, the applicant performed a general visual examination (VT-3) of the mechanical clamp type repair hardware installed around the welds instead of an enhanced visual examination (EVT-1) of the welds.

BWRVIP-18 did not require examination of repaired core spray pipe welds unless the integrity of the repair depended upon these welds. The inspectors noted that if cracks develop in the non-inspected core spray piping welds P1, 2, and 3, a cooling water flow diversion path would exist outside the core shroud which could adversely affect the applicants peak fuel clad temperature (PCT) analysis. Because the applicant's PCT analysis relied, to some extent, on the leakage integrity for these repaired welds, the inspectors determined that these welds should be inspected using EVT-1 methods to meet BWRVIP-18 requirements. Therefore, the inspectors concluded that the applicant had deviated from the BWRVIP-18 guidance and that this deviation should be identified as an exception from Section XI.M.9 of the GALL. The applicant stated that the LRA would be changed to remove statements about not inspecting these welds during the next annual LRA revision and that the applicable inspection procedures would be changed to implement enhanced visual examinations of these welds.

Based on the above enhancements being implemented, the inspectors concluded that the applicant's AMP should perform its intended function to maintain the integrity of reactor vessel internals components consistent with the current licensing basis for the period of extended operation.

13. <u>Closed-Cycle Cooling Water System (B2.1.13)</u>

The closed-cycle cooling water (CCCW) system program is an existing program which will be comparable to NUREG-1801, Section XI.M21, "Closed-Cycle Cooling Water System." However, the applicant also identified some exceptions to the GALL program. The CCCW system surveillance program manages aging effects in closed cycle cooling water systems that are not subject to significant sources of contamination, in which water chemistry is controlled and heat is not directly rejected to the ultimate heat sink. The program includes: (1) preventive measures to minimize corrosion; and (2) periodic system and component performance testing and inspection to monitor the effects of corrosion and confirm intended functions are met.

The inspectors reviewed LR program basis documentation, aging management review documents, historical chemistry parameter trends, corrective action documents, and existing procedures and surveillance. The inspectors also interviewed the CCCW program owner, interviewed the reactor building closed cooling water system engineer, and conducted walkdowns to assess the condition of CCCW systems within the plant. The inspectors verified that the applicant performed adequate historic reviews of plant specific experience to determine aging effects and the exceptions to NUREG 1801 specified in the LRA are consistent with current industry practice.

The inspectors concluded that the CCCW system program effectively manages aging effects. Continued implementation of the CCCW system program will provide reasonable assurance that the aging effects will be managed so that the CCCW system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

14. Compressed Air Monitoring (B2.1.14)

The compressed air monitoring program is an existing program that, with enhancements, will be consistent, with exceptions, to NUREG-1801, Section XI.M24, “Compressed Air Monitoring.” The MNGP compressed air monitoring program consists of inspection, monitoring, and testing of the Instrument and service air systems to provide reasonable assurance that they will perform their intended function for the duration of extended operation.

The inspectors reviewed program documentation, aging management review documents, historical chemistry parameter trends, corrective actions documents, and existing procedures. The inspectors verified that the applicant performed adequate historic reviews of plant specific experience to determine aging effects and that the exceptions to the GALL, specified in the LRA, were consistent with current industry practice.

The inspectors concluded that the compressed air monitoring program effectively manages aging effects. Continued implementation of the compressed air monitoring program will provide reasonable assurance that the aging effects will be managed so that the primary and secondary system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

15. Cables and Connections Not Subject to Environmental Qualification Requirements (B2.1.15)

The electrical cables and connectors not subject to 10 CFR 50.49 EQ requirements is a new program that, when implemented, will be consistent with the program described in NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” and guidance provided in the applicable ISGs. The electrical cables and connectors not subject to 10 CFR 50.49 EQ requirements program will apply to accessible insulated cables and connections installed in structures within the scope of LR and prone to adverse localized environments.

The inspectors reviewed aging management program related documentation, condition reports, self assessments, proposed and existing procedures, and confirmed that the applicant had NRC commitment number M05027A in place to implement the program prior to the start of the period of extended operation. The inspectors also interviewed the applicant to determine how and when the aging management program changes that are required to satisfy LR commitments and applicable ISGs will be developed and implemented. In addition, the inspectors conducted visual field inspections of electrical equipment, components and cables and verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects.

The inspectors concluded that the electrical cables and connectors not subject to 10 CFR 50.49 EQ requirements program, when implemented as described, should effectively manage aging effects, as it will incorporate proven monitoring and testing techniques, acceptance criteria, corrective actions, and administrative controls. Implementation of this program will provide reasonable assurance that the intended functions of insulated cables and connections exposed to adverse localized equipment environments caused by heat, radiation, or moisture will be maintained and that the effects of aging will be managed such that components within the scope of the program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

16. <u>Instrument Cables Not Subject to EQ Requirements (B2.1.16)</u>

The electrical cables not subject to 10 CFR 50.49 EQ requirements used in instrumentation circuits program is a new program that the applicant will implement prior to the period of extended operation. This program will be consistent with the program described in NUREG-1801, Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” with some exceptions. The electrical cables included in the scope of this program are cables which are not required to be environmentally qualified under 10 CFR 50.49; are used in radiation monitoring and nuclear instrumentation circuits with sensitive, low-level signals that are within scope of LR and are installed in adverse localized environments caused by heat, radiation and moisture in the presence of oxygen.

The inspectors reviewed aging management program related documentation, condition reports, self assessments, proposed and existing procedures, and confirmed that the applicant had NRC commitment number M05028A in place to implement the program prior to the start of the period of extended operation. The inspectors also interviewed the applicant to determine required enhancements and their implementation schedule to periodically test sensitive instrumentation circuits to ensure that the circuit will perform its intended function through the period of extended operation. The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects and that the exceptions to the GALL specified in the LRA are consistent with current industry practice and are acceptable.

The inspectors concluded that the electrical cables not subject to 10 CFR 50.49 EQ requirements used in instrumentation circuits program, when implemented as described, will effectively manage aging effects, since it will incorporate proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls. Implementation of the electrical cables not subject to 10 CFR 50.49 EQ requirements used in instrumentation circuits program will provide reasonable assurance that the effects of aging will be managed such that components within the scope of the program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

17. Fire Protection (B2.1.17)

The fire protection (FP) program is an existing program which, with the proposed enhancements, will be comparable to Section XI.M26, “Fire Protection,” of the GALL report and as clarified by ISG-04. However, the applicant also identified some exceptions to the GALL program.

The fire protection program is credited for detecting and managing age related degradation of FP system components and structures. The FP program includes fire barrier visual inspections, motor and diesel-driven fire pump tests and inspections. Periodic testing and inspection of the diesel driven fire pump is performed to ensure that an adequate flow of fire water is supplied and that there is no degradation of the diesel fuel supply lines. Fire barrier inspections will be performed, consisting of periodic visual inspection of fire barrier penetration seals, fire dampers, fire barrier walls, ceilings and floors; and periodic visual inspection and functional tests of fire-rated doors to ensure that their operability is maintained. The FP program also includes periodic inspection and testing of the Halon fire suppression system.

The inspectors reviewed fire protection aging management program related documentation, condition reports, self assessments, procedures, required enhancements, commitments and implementing documents and confirmed that the applicant had NRC commitments M05029A and M05030A in place to implement the program prior to the start of the period of extended operation. The inspectors interviewed applicant engineers to confirm the continuation of the existing program along with the implementation schedule of the required program enhancements. In addition, the inspectors verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects and the exceptions to NUREG 1801 specified in the LRA are consistent with current industry practice. The inspectors conducted field walkdowns of the fire protection system and identified one case of extensive corrosion on conductor termination lugs for the diesel driven fire pump. The applicant initiated AR 01012302 to evaluate this issue.

The inspectors concluded that the fire protection system program, in general, effectively manages aging effects. With the enhancements to be incorporated prior to the period of extended operation, continued implementation of the fire protection system program will provide reasonable assurance that the aging effects will be managed so that the fire protection system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

18. Fire Water System (B2.1.18)

The fire water system program is an existing program which, with the proposed enhancements, will be comparable to the program described in Section XI.M27, “Fire Water System,” of the GALL report, and as clarified by ISG-04.

The fire water system program consists of water-based fire protection systems that include components that are periodically inspected and tested in accordance with the applicable National Fire Protection Association codes and standards and plant procedures. These activities include sprinkler system inspections, pipe wall thickness testing, hydrant inspections, fire main flushes, and flow tests.

The inspectors reviewed fire water system aging management program related documentation, condition reports, existing procedures, required enhancements and implementing documents and confirmed that the applicant had NRC commitments M05031A, M05032A, and M05033A in place to implement the program prior to the start of the period of extended operation. The inspectors also conducted plant visual inspections to assess the condition of fire water system equipment, interviewed the fire water system engineer to confirm the continuation of the existing program along with the implementation schedule of the required program enhancements. In addition, the inspectors verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects and the exceptions to the GALL specified in the LRA are consistent with current industry practice.

The inspectors concluded that the fire water system program, in general, effectively manages aging effects. With the enhancements to be incorporated prior to the period of extended operation, continued implementation of the fire water system program will provide reasonable assurance that the aging effects will be managed so that the fire water system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

19. Flow-Accelerated Corrosion (B2.1.19)

The flow-accelerated corrosion (FAC) aging management program is an existing program credited in the LRA as being consistent with NUREG-1801, Section XI.M17, “Flow-Accelerated Corrosion.” The ongoing program is used to predict, detect, and monitor FAC in plant piping and other components, such as valve bodies, elbows, and expanders. The program was credited with: (1) conducting an analysis to determine critical locations; (2) performing baseline inspections to determine the extent of thinning at these locations; and (3) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

The inspectors reviewed the applicable LR program basis documentation, interviewed the FAC program owner, reviewed applicable procedures, reviewed the determination of systems susceptible to FAC, reviewed engineering evaluations of localized pipe wall thinning, reviewed FAC self-assessment and program health reports, reviewed a condition report and an external operating report related to FAC, and reviewed the applicant's commitment to revise the corporate fleet procedure FP-FE-FAC-01 to include 87.5 percent nominal pipe wall thickness for non-safety-related piping as a trigger for engineering analysis before the start of the period of extended operation.

The inspectors identified current plant engineering calculations for safety-related piping designed in accordance with the ASME Code that used an alternative method to evaluate localized pipe wall thinning instead of ASME Code Case 597-1. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," lists Code case 597-1 as a conditionally acceptable code case. The applicant committed to evaluate the 2003 alternative methodology, address the use of Code Case 597-1 in future calculations, and review the relationship between the FAC program and ASME Section XI requirements when evaluating wall thinning in AR 01013831.

The inspectors concluded that the FAC program was in place, had been implemented, was an ongoing program subject to NRC review, and generally included the elements identified in the LRA. As it is a current program subject to periodic NRC review and inspection, there is reasonable assurance that adequate inspections required by the program will be performed through the period of extended operation.

20. Fuel Oil Chemistry (B2.1.20)

The fuel oil chemistry program is an existing program that, with enhancements, will be consistent, with exceptions, to NUREG-1801, Section XI.M30, "Fuel Oil Chemistry." The fuel oil program mitigates and manages aging effects on the internal surfaces of diesel fuel oil storage tanks and associated components in systems that contain diesel fuel oil. The program includes: a) surveillance and monitoring procedures for maintaining diesel fuel oil quality by controlling contaminants in accordance with applicable ASTM Standards; b) periodic draining of water from diesel fuel oil tanks, if water is present; c) periodic or conditional visual inspection of internal surfaces or wall thickness measurements from external surfaces of diesel fuel oil tanks; and d) one-time inspections of a representative sample of components in systems that contain diesel fuel oil.

The inspectors reviewed LR program basis documentation, aging management review documents, and existing procedures. The inspectors also interviewed the program owner and conducted walkdowns of the emergency diesel generators, day tanks, and enclosures for the fuel oil storage tanks. The inspectors confirmed that the applicant had commitments in place to enhance the program prior to the start of the period of extended operation. The inspectors verified that the applicant performed adequate historic reviews of plant specific experience to determine aging effects. The inspectors confirmed that the exception to the GALL to not require addition of biocides, stabilizers, and corrosion inhibitors was acceptable based on plant history and on the continuing requirement to sample the fuel oil and evaluate abnormal test results.

The inspectors concluded that the fuel oil chemistry program effectively manages aging effects. Providing the enhancements are incorporated as specified by the applicant's application, continued implementation of the fuel program will provide reasonable assurance that the aging effects will be managed so that the fuel oil system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

21. Inaccessible Medium Voltage Cables Not Subject to EQ (B2.1.21)

The inaccessible medium-voltage cables not subject to 10 CFR 50.49 requirements program is a new program that the applicant will implement prior to the period of extended operation. The program, when implemented will be consistent with that described in NUREG-1801, Section XI.E3, "Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Requirements." This program applies to inaccessible (e.g., in conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct-buried) medium-voltage cables within the scope of LR that are exposed to significant moisture simultaneously with applied voltage. This program includes a commitment to periodically inspect and test these cables to provide an indication of the condition of the conductor insulation. The specific type of tests performed will be determined and implemented prior to the expiration of the current license.

The inspectors reviewed aging management program related documentation, condition reports, existing procedures, and conducted field inspections of a number of manholes in the plant and in the substation and confirmed that the applicant had NRC commitment M05037A in place to implement the program prior to the start of the period of extended operation. The inspectors also interviewed applicant engineers to determine, in general, how the program would be enhanced to include manhole inspections and additional "state of the art" cable tests. In addition, the inspectors verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects.

The inspectors concluded that the inaccessible medium-voltage cables not subject to 10 CFR 50.49 requirements program, when implemented as described in the required enhancements, will effectively manage aging effects, since it will incorporate periodic inspections and "state of the art" testing techniques. Implementation of this program will provide reasonable assurance that the effects of aging will be managed such that components within the scope of the program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

22. Inspection of Overhead Heavy and Light Load Handling Systems (B2.1.22)

The inspection of overhead heavy load and light load (related to refueling) handling systems program is an existing program which, when enhanced, will be comparable to NUREG-1801, Section XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems." However, the applicant also identified an exception to the GALL program. The purpose of overhead heavy load and light load (related to refueling) handling systems program is to identify component aging effects prior to loss of intended function. The program manages aging effects of the structural components for the cranes, heavy loads, rigging (reactor handling equipment) systems, and structures. Crane rails and structural components are visually inspected for indications of degradation, such as corrosion, wear, or cracks.

The inspectors reviewed the applicable LR program basis documentation, existing overhead crane inspection procedures, and confirmed that the applicant had commitments in place to enhance the program to specify a five-year inspection frequency for the fuel preparation machines prior to the start of the period of extended operation. The inspectors also interviewed the crane program owner and maintenance personnel that perform overhead crane structural inspections, reviewed documentation associated with "engineered lifts", and reviewed condition reports to verify identified crane structural concerns are being addressed through the applicant's corrective action program.

The inspectors reviewed the applicant's exception to a statement in Section XI.M23 of the GALL report. The applicant took exception to tracking the number and magnitude of lifts made by the crane, as the current program does not track the number and magnitude of lifts within the crane's rated capacity. The inspectors determined the applicant does track the number and magnitude of "engineered lifts" that exceed the crane's rated capacity, and these lifts are controlled by applicant procedures. Therefore, the inspectors concluded that this exception was acceptable.

The inspectors concluded that the inspection of overhead heavy load and light load handling systems program effectively manages aging effects. With the enhancement to be incorporated prior to the period of extended operation, continued implementation of the inspection of overhead heavy load and light load (related to refueling) handling systems program will provide reasonable assurance that the aging effects will be managed so that the monitored structural components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

23. One-Time Inspection Program (B2.1.23)

The one-time inspection program is a new program that will be generally consistent with NUREG-1801, Section XI.M32, "One-Time Inspection." The program will include measures to verify the effectiveness of the plant and fuel oil chemistry aging management programs. Additionally, the program will confirm the absence of age degradation in selected components such as flow restrictors, venturis, and small bore piping that are within the scope of license renewal. The program is to be implemented prior to the period of extended operation.

The inspectors reviewed the applicant's LR documentation and the one-time inspection program sampling method implementation procedure. The inspectors also discussed the planned scope and methodology for the program with applicant engineers and confirmed that the applicant had an existing commitment to implement the program prior to the period of license renewal. The inspectors concluded that, provided the program was implemented in accordance with the commitment and planned methodology, the program should provide reasonable assurance that the effects of aging would be assessed and managed such that components within the scope of the program will perform their intended functions consistent with the current licensing basis for the period of extended operation

24. Open-Cycle Cooling Water System (B2.1.24)

The open-cycle cooling water system program is an existing program which is comparable to NUREG-1801, Section XI.M20, "Open-Cycle Cooling Water System." The open-cycle cooling water system program to ensure that the effects of aging on the raw water service water systems will be managed for the period of extended operation. This program manages the aging effects of metallic components in water systems exposed to raw, untreated water. These aging effects are due to corrosion, erosion, and biofouling in systems, structures and components serviced by the OCCW system. The program includes: a) surveillance and control of biofouling; b) tests to verify heat transfer; and c) routine inspection and maintenance.

The inspectors reviewed program documentation, aging management review documents, corrective actions documents, and existing procedures and surveillance. The inspectors also interviewed the program owners, interviewed the service water system engineer and conducted walkdowns to assess the condition of the service water system. The inspectors verified that the applicant performed adequate historic reviews of plant specific experience to determine aging effects.

During an OCCW system walkdown, the inspectors identified floor drain pipes that did not appear on the LR boundary drawings and that were within the scope of license renewal. These drains fell within the wells and domestic water system scoping and screening program. The applicant committed to revise the LRA and supporting documents to accommodate these drain lines as being within scope of license renewal.

The inspectors concluded that the OCCW system program effectively manages aging effects. Continued implementation of the OCCW system program will provide reasonable assurance that the aging effects will be managed so that the service water system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

25. Plant Chemistry Program (B2.1.25)

The water chemistry program is an existing program that is consistent with NUREG-1801, Section XI.M2, "Water Chemistry." However, the applicant also identified some exceptions to the GALL program. The plant chemistry program mitigates the aging effects on component surfaces that are exposed to water as the process fluid; chemistry programs are used to control water chemistry for impurities (e.g., chloride and sulfate) that accelerate corrosion or crack initiation and growth and that cause heat transfer degradation due to fouling in select heat exchangers.

The inspectors reviewed program documentation, aging management review documents, historical chemistry parameter trends, corrective actions documents, and existing procedures. The inspectors verified that the applicant performed adequate historic reviews of plant specific experience to determine aging effects and that the exceptions to NUREG 1801 specified in the LRA are consistent with current industry practice.

The inspectors concluded that the water chemistry program effectively manages aging effects. Continued implementation of the water chemistry program will provide reasonable assurance that the aging effects will be managed so that the primary and secondary system components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

26. <u>Primary Containment Inservice Inspection Program (B2.1.26)</u>

The primary containment ISI program is an existing program which is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE.” The primary containment ISI program requires visual examinations of the accessible surfaces (base metal and welds) of the drywell, torus, vent lines, internal vent system, penetration assemblies and associated integral attachments. The program also requires examination of pressure retaining bolting and the drywell interior slab moisture barrier. The primary containment ISI program manages the aging effects of corrosion, cracking, mechanical damage, discoloration and other phenomena that can potentially impact structural and/or leak tight integrity. Most examination surfaces are coated; therefore, examinations focus on coating conditions that indicate the possible deterioration of the underlying metal.

The inspectors reviewed LR program basis documentation, aging management review documents, existing plant procedures, and recently completed inspection results. The inspectors verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects.

The inspectors concluded that the primary containment ISI program effectively manages aging effects. Continued implementation of the primary containment ISI program will provide reasonable assurance that the aging effects will be managed so that the structural components within the scope of the program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

27. <u>Protective Coating Monitoring and Maintenance Program (B2.1.27)</u>

The protective coating monitoring and maintenance program is an existing program which, when enhanced, will be consistent with the recommendations of

NUREG-1801, Section XI.S8, “Protective Coating Monitoring and Maintenance Program.” The program provides for inspections of the drywell and torus interior coated surfaces. The torus inspections include both above- and below-water line surface inspections. The protective coating monitoring and maintenance program is not relied upon to manage loss of material due to corrosion of carbon steel but, rather, is primarily relied upon to ensure that degradation of protective coatings inside containment won't lead to clogging of the emergency core cooling system (ECCS) suction strainers.

The protective coating monitoring and maintenance program requires inspections to be performed each operating cycle for the torus above-water line and drywell coated surfaces. The program further states that below-water line painted surfaces, including the areas near the ECCS suction strainers, are inspected at intervals not exceeding five years and that a VT-3 exam is performed for all coating degradation identified. All unacceptable areas are required to be repaired or evaluated.

During a review of the applicant’s procedures 0135, “Pressure-Suppression Chamber Painted Surface Internal Inspection,” and 0140, “Drywell Interior Surface Inspection,” the inspectors identified that, for the previous two performances of the inspections, the applicant removed the step which required VT-3 examination of unacceptable areas because the examination was not a Code requirement. The applicant issued AR 00830109-03 to revise the procedures to ensure the requirement to perform VT-3 examinations for coatings is implemented to meet LR aging management program criteria.

In addition, the inspectors reviewed procedure 1367, “Pressure-Suppression Chamber Below Water Line Painted Surface Internal Inspection.” The purpose section of the procedure stated that inspections were to be conducted at internals not to exceed five years. However, in another section, the frequency is stated as “approximately every (5) years.” The inspectors noted a six-year interval between the previous two inspections. The applicant issued AR 00830109-02 to revise procedure 1367 to reflect the maximum surveillance frequency of five years in accordance with the aging management program.

The inspectors reviewed LR program basis documentation, aging management review documents, existing plant procedures, and recently completed inspection results. The inspectors verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects.

The inspectors concluded that the protective coating monitoring and maintenance program effectively manages aging effects. Continued implementation of the protective coating monitoring and maintenance program will provide reasonable assurance that the aging effects will be managed so that the structural components within the scope of the program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

28. Reactor Head Closure Studs (B2.1.28)

The reactor head closure studs aging management program is an existing program credited in the LRA which, when enhanced, will be comparable to NUREG-1801, Section XI.M3, “Reactor Head Closure Studs.” The scope of the ongoing program is part of the ASME Section XI inservice inspection program and provides condition monitoring of the reactor head closure studs, nuts, threads in flange, closure washers, and bushings. Preventative measures to mitigate cracking have been taken in accordance with Regulatory Guide (RG) 1.65, “Materials and Inspections for Reactor Vessel Closure Studs.”

The inspectors reviewed the applicable LR program basis documentation, interviewed the reactor head closure studs program owner, inspected a sample of reports for ultrasonic examination of the reactor head closure studs, reviewed the applicant’s evaluation of external operating experience related to reactor head closure stud cracking, and reviewed applicable program procedures.

The inspectors noted that RG 1.65 recommended that reactor head closure stud material measured ultimate tensile strength not exceed 170 ksi [kilo-pounds per square inch] to minimize the likelihood of stress corrosion cracking. Hardness tests conducted on the installed reactor head studs showed that most studs to have greater than 170 ksi tensile stress. The applicant committed to document this exception to NUREG-1801 in the LRA. The exception will state that the aging management program does not incorporate the tensile strength requirement of RG 1.65 for existing reactor head closure studs. The inspectors concluded that this exception to NUREG-1801 was acceptable because the applicant considers these studs to be susceptible to cracking, continues to manage the studs using the other preventative measures of RG 1.65, continues to conduct ultrasonic testing and surface examinations on a ten year interval, and to date no apparent discontinuities have been identified.

The inspectors concluded that inspection reactor head closure studs was part of the ISI – ASME Section XI program, the program had been implemented, was an ongoing program subject to NRC review, and included the elements identified in the LRA. As it is a currently required program subject to periodic NRC review and inspection, there is reasonable assurance that adequate inspections required by ASME and the NRC will be performed through the period of extended operation.

29. Reactor Vessel Surveillance (B2.1.29)

The reactor vessel surveillance program is an existing program which is part of the BWRVIP integrated surveillance program (ISP) that uses data from BWR member surveillance programs to select the “best” representative vessel materials to monitor radiation embrittlement for a particular plant. The applicant identified that the reactor vessel surveillance program would be enhanced to be consistent with the recommendations of NUREG-1801, Chapter XI, Program XI.M.31 “Reactor Vessel Surveillance.” The AMP enhancement was the applicant’s commitment to follow BWRVIP-116 “BWR Vessel and Internals

Project, Integrated Surveillance Program Implementation for License Renewal" after NRC approval. Because the NRC review of BWRVIP-116 was ongoing, the inspectors did not assess the program for compliance with this enhancement. In NRC audit and review report for plant aging management reviews and programs (ML052850461), the NRC stated that this program would be reviewed by the NRR staff and issues addressed within Section 3 of the SE for the applicant's LRA.

The inspectors reviewed LR program basis documents associated with the applicants reactor vessel surveillance program and the calculation which demonstrated that the vessel would maintain adequate toughness (equivalent margins for upper shelf impact energy) to meet 10 CFR Part 50, Appendix G, requirements for the planned 60 year service life. The inspectors noted that the equivalent margins calculation supported the information for the limiting vessel materials identified in Table 4.2.1-1 "Equivalent Margin Analysis for MNGP Plate Material," of the LRA. The inspectors also searched the applicant's corrective action program records for issues related to reactor vessel surveillance to assess the program's effectiveness at detecting and monitoring for age related degradation (e.g., vessel embrittlement). Based upon these reviews, the inspectors did not identify any issues adversely affecting the applicant's AMP. Therefore, the reactor vessel surveillance program should continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

The inspectors identified a current plant issue associated with the applicant's planned change to their vessel surveillance capsule withdraw schedule. In BWRVIP-86-A "BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program Implementation Plan," the ISP schedule defined that a surveillance capsule specimen was due to be removed from Monticello in 2006. The inspectors noted that the next Monticello surveillance capsule sample was scheduled to be withdrawn during the spring 2007 Monticello refueling outage. This change in schedule was accepted by the BWRVIP, but had not been submitted and approved by the NRC as required by paragraph III.B.3 of 10 CFR Part 50, Appendix H. The applicant intended to work with the BWRVIP to address this change in schedule and to ensure that the schedule change was approved by the NRC.

30. <u>Selective Leaching of Materials (B2.1.30)</u>

The selective leaching of materials inspection activities is a new program which, when implemented, will be consistent, with exceptions, to Section XI.M33, "Selective Leaching of Materials," of the GALL report. The program will consist of one-time visual inspection and hardness measurement of selected components that are susceptible to selective leaching. In situations where hardness testing is not practical, a qualitative method by non-destructive examination or other metallurgical methods will be used to determine the presence and extent of selective leaching.

The inspectors reviewed the LR evaluation, interviewed the selective leaching of materials program owner, and reviewed the applicant's commitment to develop and implement a selective leaching of materials inspection program before the start of the period of extended operation. The inspectors concluded that, if the procedures are implemented as planned, there should be reasonable assurance that aging effects will be managed so that components susceptible to selective leaching of materials will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

31. <u>Structures Monitoring Program (B2.1.31)</u>

The structural monitoring program is an existing program which, when enhanced, will be comparable to NUREG-1801, Sections XI.S5, "Masonry Wall Program," XI.S6, "Structures Monitoring Program," and XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants." The structural monitoring program manages aging effects to ensure that structures and components within its scope retain the ability to perform their intended function and is implemented through visual examination of the structures and components. The program is implemented as part of the structures monitoring performed under the provisions of the Maintenance Rule, 10 CFR 50.65, with additional inspections of the intake structure and diesel fuel oil transfer house. The program is based on guidance contained in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The program, when enhanced, will include the monitoring and inspection parameters for structural components within the scope of license renewal.

The inspectors reviewed the applicable LR program basis documentation, existing structural monitoring inspection procedures, and confirmed that the applicant had commitments in place to enhance the program prior to the start of the period of extended operation. The inspectors also interviewed the structural monitoring program owner, performed walkdowns of selected structures, reviewed a maintenance rule program health status report, a periodic assessment report, a structural monitoring inspection report, and various deficiency reports, condition reports, and external operating experience to verify identified structural concerns are being evaluated and corrected if necessary.

The inspectors concluded that the structural monitoring inspection program effectively manages aging effects. When enhanced as described in the application, continued implementation of the structural monitoring inspection program will provide reasonable assurance that the aging effects will be managed so that the monitored components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

32. <u>System Condition Monitoring Program (B2.1.32)</u>

The system monitoring program is an existing plant-specific program which manages aging effects for normally accessible, external surfaces of piping, tanks, and other components and equipment within the scope of license renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation. The program relies upon periodic system walkdowns to monitor degradation of the protective paint or coating, and/or the exterior steel surface area (if no paint or coatings exist, or if the existing protective paint and coatings are degraded to a point whereby the exterior steel surface is exposed). Although no credit is taken for any coating or paint, inspections of the above-ground coating or paint will provide an indication of the condition of the material underneath the coating or paint. The program will monitor for loose, worn or missing parts, fluid leaks, bolting or fastener degradation, and evidence of corrosion and sealant degradation. The minimum walkdown frequency is once per year for those systems and components that are accessible during normal operation. Current walkdowns are conducted every four months. The inspection frequency may be increased based on the safety significance, production significance, or operating experience of each system. Systems and components that are only accessible during plant outages are inspected at least once per refueling interval. The applicant planned to enhance the current system walkdown procedure to describe specific age degradation parameters to be monitored and inspected. Acceptance criteria will also be included.

The inspectors reviewed LR program basis documentation, aging management review documents, and the existing system engineering walkdown procedure. The inspectors also accompanied the reactor building closed cooling water system engineer on a routine system walkdown. The inspectors verified that the applicant performed adequate historic reviews of plant specific experience to determine aging effects.

The inspectors concluded that the system monitoring program adequately managed current plant aging effects. With the enhancements to be incorporated prior to the period of extended operation, continued implementation of the system monitoring program will provide reasonable assurance that the aging effects will be managed so that the plant components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

33. <u>Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (B2.1.33)</u>

The thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) program is an existing program that is consistent with NUREG 1801, Section XI.M.13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel” with one exception as described in NRC audit and review report for plant aging management reviews and programs (ML052850461). The program provides for condition monitoring of the CASS components subject to thermal or irradiation induced embrittlement.

The inspectors reviewed LR program basis documents and searched the applicant's corrective action program records for issues related to CASS components to assess the program's effectiveness at detecting and monitoring for degraded CASS components. The applicant concluded that the CASS components (e.g., fuel support pieces) within the core which met the screening criteria for potentially being susceptible to neutron irradiation induced embrittlement did not require augmented inspections due to compressive loadings. The inspectors noted that this position was consistent with NUREG 1801 Section XI.M.13 recommendations.

The inspectors did not identify any issues adversely affecting the applicant's AMP. The inspectors also did not identify any additional exceptions from the Section X1.M.13 program. Therefore, the applicant's AMP should continue to perform its intended function and maintain the integrity of CASS components consistent with the current licensing basis for the period of extended operation.

34. <u>Environmental Qualification (B3.1)</u>

The electrical equipment qualification program is an existing ongoing program which manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished or replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered time-limited aging analyses (TLAA) for license renewal. The EQ program ensures that these EQ components are maintained within the bounds of their qualification bases. The program is consistent with NUREG-1801, Section X.E1, "Environmental Qualification of Electric Components."

The inspectors reviewed aging management program related documentation, condition reports, self assessments and existing procedures to confirm that the applicant has been successful in effectively managing aging effects of EQ electric components. The inspectors also interviewed the EQ program owners to confirm that the applicant will continue to carry out the EQ program for the duration of the extended operation. The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects.

The inspectors concluded that the applicant's existing EQ program has been effective in managing aging effects. The program has been subject to periodic internal and external assessments that facilitate continuous improvement. With continued implementation and effective management, the EQ program will provide reasonable assurance that the aging effects will be managed so that the environmentally qualified plant components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

35. Metal Fatigue of the Reactor Coolant Pressure Boundary (B3.2)

The fatigue monitoring program is an existing program which, when enhanced, will be comparable to Section X.M1, “Metal Fatigue of the Reactor Coolant Pressure Boundary,” of the GALL report. The fatigue monitoring program is a confirmatory program that monitors loading cycles due to thermal and pressure transients and cumulative fatigue usage for selected reactor coolant and other component locations. Metal fatigue analyses is considered a TLAA under license renewal. The program provides an analytical basis for confirming that the actual number of cycles does not exceed the number of cycles used in the design analysis and that the cumulative usage is maintained below the allowable limit, or that appropriate corrective actions are taken to maintain component cumulative fatigue usage below the allowable limit during the period of extended operation. Enhancements include the incorporation of fatigue sensitive locations for older vintage GE plants identified in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves of Selected Nuclear Power Plant Components,” into the fatigue monitoring program implementing procedures.

The inspectors reviewed the applicable LR program basis documentation, interviewed the fatigue monitoring program owner, reviewed applicable procedures, confirmed commitments are in place to include the identified enhancement into implementing procedures, reviewed a thermal fatigue program health status report, reviewed a sample of fatigue monitoring analytical calculations to confirm that the evaluations included the period of extended operation and the effects of reactor water environment if applicable, and reviewed applicant corrective actions of external operating experience related to the fatigue monitoring program.

The inspectors concluded that the fatigue monitoring program effectively manages aging effects. When enhanced as described, continued implementation of this program will provide reasonable assurance that the aging effects will be managed so that the monitored components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

E. Exit Meeting Summary

The results of this inspection were discussed on February 22, 2005, with Mr. R. Jacobs, and other members of the Nuclear Management Corporation staff in an exit meeting open for public observation at the Monticello Community Center in Monticello, Minnesota. The applicant acknowledged the inspection results and presented no dissenting comments. The slides used during this meeting are provided in the supplemental information.

The inspectors noted that proprietary documents were reviewed during the course of the inspection. The applicant confirmed that all such proprietary documents were returned or the copies destroyed and that the likely content of the report would not involve the proprietary material.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Applicant
M. Aleksey, License Renewal TLAA Lead
R. Baumer, Licensing
P. Burke, Manager of Projects
R. Dennis, License Renewal Civil Lead
J. Grubb, Engineering Director
R. Jacobs, Site Director for Operations
J. Pairitz, License Renewal Project Manager and Mechanical Lead
J. Rootes, License Renewal Programs Lead
B. Sawatzke, Plant Manager
R. Siepel, License Renewal Electrical Lead

Nuclear Regulatory Commission
A. Stone, Chief, Engineering Branch 2
B. Burgess, Chief, DRP Branch 4
S. Thomas, Senior Resident Inspector
R. Orlikowski, Resident Inspector

LIST OF DOCUMENTS REVIEWED

The following is a list of applicant documents reviewed during the inspection, including documents prepared by others for the applicant. Inclusion of a document on this list does not imply that NRC inspectors reviewed the entire documents, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. In addition, inclusion of a document on this list does not imply NRC acceptance of the document, unless specifically stated in the body of the inspection report.

LICENSE RENEWAL DOCUMENTS

License Renewal Application

- Application for Renewed Operating License; March 2005

License Renewal Action Items (Written as a Result of the Inspection)

- 00829856-02; Add Base Plate Gap Inspection Requirement and Acceptance Criteria to Implementing Procedure PEI-02.05.02; dated February 3, 2006
- 00830392-02; Add Base Plate Gap Inspection Requirement and Acceptance Criteria to System Walkdown Guidelines; dated February 3, 2006
- 00829888-02; Sub Task to Track Detailed Inspection Period Requirements for Bus Duct; dated February 3, 2006
- 00830109-02; Revise Procedure 1367 to Specify 5-Year Surveillance Frequency; dated February 9, 2006
- 00830109-03; Ensure VT-3 Exams Remain in Protective Coatings Procedures; dated February 9, 2006
- 00874822-02; Evaluate Non-Safety Components < 87.5 percent T_{nom} Attributable to Flow Accelerated Corrosion; dated February 8, 2006

License Renewal Aging Management Program Basis Documents

- PBD/AMP-001; Plant Chemistry Program; Revision 1
- PBD/AMP-002; Flow Accelerated Corrosion Program; Revision 2
- PBD/AMP-004; Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Program; Revision 1
- PBD/AMP-006; Bolting Integrity; Revision 2
- PBD/AMP-007; Open Cycle Cooling Water System Program; Revision 1
- PBD/AMP-008; Closed Cycle Cooing Water Program; Revision 1
- PBD/AMP-010; Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems; Revision 1
- PBD/AMP-011; Compressed Air Monitoring Program; Revision 1
- PBD/AMP-013; Fire Protection; Revision 1
- PBD/AMP-014; Fire Water System; Revision 1
- PBD/AMP-017; Fuel Oil Chemistry; Revision 1
- PBD/AMP-018; Reactor Vessel Surveillance; Revision 1
- PBD/AMP-019; One-Time Inspection; Revision 1
- PBD/AMP-020; Selective Leaching of Materials; Revision 1
- PBD/AMP-021; Buried Piping and Tanks Inspection; Revision 1
- PBD/AMP-022; Primary Containment In-Service Inspection Program; Revision 1

- PBD/AMP-024; ASME Subsection XI, Subsection IWF; Revision 1
- PBD/AMP-025; 10 CFR 50, Appendix J Program; Revision 2
- PBD/AMP-027; Structures Monitoring Program; Revision 1
- PBD/AMP-029; Protective Coating Monitoring and Maintenance Program; Revision 1
- PBD/AMP-030; Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification (EQ) Requirements; Revision 1
- PBD/AMP-031; Electrical Cables Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits; Revision 1
- PBD/AMP-031; Boiling Water Reactor (BWR) Penetrations; Revision 1
- PBD/AMP-032; Inaccessible Medium Voltage (2kV to 34.5kV) Cables Not Subject to 10 CFR 50.49 EQ Requirements; Revision 1
- PBD/AMP-033; ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD; dated February 18, 2005
- PBD/AMP-034; Reactor Head Closure Studs; Revision 1
- PBD/AMP-035; BWR Inside Diameter Attachment Welds; Revision 1
- PBD/AMP-036; BWR Feedwater Nozzle; Revision 1
- PBD/AMP-037; BWR Control Rod Drive Return Nozzle; Revision 1
- PBD/AMP-038; BWR Stress Corrosion Cracking; Revision 2
- PBD/AMP-040; BWR Vessel Internals; Revision 1
- PBD/AMP-042; Electrical Equipment Subject to 10 CFR 50.49 EQ Requirements; Revision 1
- PBD/AMP-043; Metal Fatigue of the Reactor Coolant Pressure Boundary; Revision 1
- PBD/AMP-044; System Condition Monitoring Program; Revision 0
- PBD/AMP-045; Bus Duct Inspection Program; Revision 2

<u>License Renewal Aging Management Review Reports</u>

- AMR-AIR; Instrument and Service Air System; Revision 1
- AMR-CABLES; Non-EQ Cables and Connectors Commodity Group; Revision 3
- AMR-CDR; Main Condenser System; Revision 1
- AMR-CFW; Condensate and Feedwater System; Revision 1
- AMR-DGN; Emergency Diesel Generator System; Revision 1
- AMR-FPC; Fuel Pool Cooling and Cleanup System; Revision 0
- AMR-FUSEHOLD; Fuse Holders Commodity Group; Revision 1
- AMR-HTV; Heating and Ventilation System; Revision 3
- AMR-RAD; Liquid and Solid Radwaste System; Revision 1
- AMR-REC; Reactor Recirculation System; Revision 2
- AMR-RWC; Reactor Water Cleanup System; Revision 1
- AMR-SLC; Standby Liquid Control System; Revision 2
- AMR-SSW; Service and Seal Water System; Revision 1

<u>License Renewal Drawings</u>

- LR-119259; Zinc Injection Passivation System; Revision 0
- LR-155483-1; Wells and Domestic Water; Revision 0
- LR-36033; Main Steam; Revision 1
- LR-36034; Turbine and Extraction Steam, Sheet 1; Revision 2
- LR-36035; Turbine and Extraction Steam; Revision 1
- LR-36035-2; Steam Jet Air Ejectors; Revision 1
- LR-36036; Condensate and Feedwater; Revision 1

- LR-36037; Condensate and Feedwater, Revision 1
- LR-36038; Condensate Demineralizer System; Revision 0
- LR-36038-2; Condensate Demineralizer System; Revision 0
- LR-36039; Condensate and Demineralized Water Storage Systems; Revision 2
- LR-36041; Service Water System, Sheet 1; Revision 1
- LR-36041-2; Service Water System Sheet 2; Revision 1
- LR-36042; Reactor Building Cooling Water System; Revision 1
- LR-36043; Radwaste Sump System Open (Dirty); Revision 1
- LR-36044; Radwaste Sump System Closed (Clean); Revision 1
- LR-36045; Clean Radwaste System; Revision 1
- LR-36046; Dirty Radwaste System; Revision 1
- LR-36047-1; Radwaste Solids Handling System; Revision 1
- LR-36048; Fire Protection System Interior Locations; Revision 1
- LR-36049-1; Compressed Air System; Revision 0
- LR-36049-4; Service Air System; Revision 1
- LR-36049-10; Alternate Nitrogen Supply System; Revision 1
- LR-36049-12; Reactor Building and Drywell Instrument Air; Revision 1
- LR-36049-13; Instrument Air–Reactor Building; Revision 1
- LR-36049-14; Instrument Air–Reactor Building; Revision 1
- LR-36051; Diesel Oil System, Sheet 1; Revision 1
- LR-36051-1; Diesel Fuel Oil System, Sheet 2; Revision 1
- LR-36241; Nuclear Boiler System Steam Supply; Revision 1
- LR-36242; Vessel Instrumentation Nuclear Boiler System; Revision 1
- LR-36254; Reactor Water Cleanup System; Revision 1
- LR-36256; Fuel Pool Cooling and Cleanup System; Revision 2
- LR-36259; Auxiliary and Heating Steam System, Turbine Building; Revision 1
- LR-36259-1; Auxiliary and Heating Steam System, Miscellaneous Areas; Revision 1
- LR-36260; Auxiliary and Heating Steam System, Reactor and Radwaste Buildings; Revision 1
- LR-36261; Chilled Water Piping and Miscellaneous, Reactor and Radwaste Building; Revision 0
- LR-36263; Air Flow Diagram, Turbine Building; Revision 1
- LR-36266; Air Flow Diagram, Radwaste Area; Revision 1
- LR-36267; Plant Air Flow Diagram, Sheet 1; Revision 1
- LR-36298; Electrical Load Flow One Line Diagram; Revision 2
- LR-36348; Auxiliary Heating System Heating Boiler; Revision 1
- LR-36516; Fire Protection System Yard Areas; Revision 1
- LR-36664; Residual Heat Removal Service Water and Emergency Service Water Systems; Revision 2
- LR-36665; Service Water System and Makeup Intake Structure; Revision 1
- LR-36665-2; Service Water System and Makeup Intake Structure; Revision 1
- LR-36666; Screen Wash, Fire and Chlorination System Intake Structure; Revision 1
- LR-36776; Control Diagram, Intake and Discharge Structures; Revision 1
- LR-36807; Airflow Diagram Reactor Building Lower Part; Revision 1
- LR-46162; Primary Containment Nitrogen Control System; Revision 1

<u>License Renewal Miscellaneous Documents</u>

- GE-NE-0000-0020-0279-01; Time-Limited Aging Analyses–Reactor Vessel and Internals; dated April 2004

- LRPP 1-5; License Renewal Implementation Plan; Revision DRAFT
- LRPP 1-3; Operating Experience Data Collection; dated June 29, 2004
- Meeting Notes; License Renewal One Time Inspection Expert Panel; dated December 8, 2005
- One Time Inspection Program Sampling Method; dated February 9, 2006

License Renewal Scoping and Screening Reports

- SSR-AIR; Instrument and Service Air System; Revision 1
- SSR-CDR; Main Condenser; Revision 1
- SSR-CFW; Condensate and Feedwater System; Revision 1
- SSR-DAC; Drywell Atmospheric Cooling System; Revision 0
- SSR-DGN; Emergency Diesel Generators; Revision 4
- SSR-FPC; Fuel Pool Cooling and Cleanup System; Revision 2
- SSR-HTV; Heating and Ventilation System; Revision 2
- SSR-INT; Reactor Pressure Vessel Internals; Revision 1
- SSR-NDG; Non-Essential Diesel Generator System; Revision 1
- SSR-RAD; Radwaste Solid and Liquid System; Revision 1
- SSR-RWC; Reactor Water Cleanup System; Revision 1
- SSR-SSW; Service and Seal Water System; Revision 1

License Renewal Operating Experience Review Output Reports

- OE-CDR; Main Condenser System; Revision 0
- OE-CFW; Condensate and Feedwater System; Revision 0
- OE-FPC; Fuel Pool Cooling and Cleanup System; Revision 0
- OE-HTV; Heating and Ventilation System; Revision 0

License Renewal Technical Reports

- TR-003; Component Identification and Data Processing for Systems, Structures and Components (SSC) Within Scope of 10 CFR 54.4(a)(3) for Environmental Qualification; Revision 1
- TR-004; Component Identification for SSC within Scope of 10 CFR 54.4(a)(3) for Fire Protection Program; Revision 2

CURRENT PLANT DOCUMENTS

Calculations

- CA-01-116; Documentation of Thermal Cycles, including Addendum 1; Revision 0
- CA-03-163; Determination of Remaining Allowable Operating Time for TW34-10"-HE (Code); Revision 0
- CA-03-170; Evaluation of TW24-10"/12"-HE in Corroded Condition; Revision 0
- CA-04-142; Environmental Fatigue Calculation; Revision 0
- CA-04-143; Environmental Fatigue Calculations for NUREG/CR-6260 Locations; Revision 0
- CA-04-166; 12 Emergency Diesel Generator Essential Service Water HX Performance Test–Summer 2005; Revision 0
- CA-96-086; 1996 Core Shroud Evaluation Analysis; Revision 7

Corrective Action Requests (AR) Initiated As a Result of the Inspection

- 01011589; Cracks in High Pressure Coolant Injection Pedestal Grout Cap; dated January 24, 2006
- 01012112; Degraded Components Not Expeditiously Repaired or Replaced; dated January 27, 2006
- 01012190; Questionable Note in Station Blackout Procedure; dated January 27, 2006
- 01012302; Division 1 Emergency Diesel Generator C-91 Panel Wire Bundle Discoloration; dated January 28, 2006
- 01012303; #12 Emergency Diesel Generator Maintenance Concerns; dated January 28, 2006
- 01012304; 1R Metal Enclosed Bus Duct Cover Bent; dated January 28, 2006
- 01013177; Procedure Change Request Past Due
- 01013360; Frequency of 1R Bus Inspection Does Not Meet Equipment Excellence–Was Last Performed in 1994; dated February 3, 2006
- 01013447; Swallows Nest on 1R Transformer and Duct Support; dated January 25, 2006
- 01013635; Unreliable 1R Trouble Annunciator (Hot Oil Portion); dated February 6, 2006
- 01013672; Noted Previous Leakage from Wall by Torus 896'; dated February 6, 2006
- 01013829; Swallows Nest on 2R Transformer and Duct Support; dated January 25, 2006
- 01013831; Calculation Method for Wall Thinning on Residual Heat Removal Piping in 2003; dated February 7, 2006
- 01013858; Small Diesel Fuel Oil Leak From Diesel Fire Pump Engine; dated February 7, 2006
- 01013874; Rust on Cathodic Protection Junction Box; dated February 7, 2006
- 01013875; Six Limited In-Service Inspection Exams not Included in Cycle 19 Relief Requests; dated February 7, 2006
- 01013930; Seal Water Pressure Gauges Swinging ±10 Psi; dated February 7, 2006
- 01014163; Intake Structure Ceiling Penetrations Material Condition; dated February 9, 2006

Corrective Action Documents Reviewed During the Inspection

- 000032; Self Assessment from Vermont Yankee for Steam Dryer; dated May 4, 2004
- 000137; Apparent Cause Evaluation: Crack Indication on Control Rod Drive Withdrawal Line Found in Drywell
- 000173; Condition Evaluation: Flow Accelerated Corrosion Program Fleet Procedure Issue; dated March 28, 2005
- 000298; Root Cause Evaluation: Possible Cracked Control Rod Drive Withdrawal Line in Drywell
- 000512; Corrective Action: Flow Accelerated Corrosion Program Fleet Procedure Issue; dated May 20, 2005
- 000532; Flow Accelerated Corrosion Program Fleet Procedure Issue; dated March 24, 2005
- 001952; Apparent Cause Evaluation: Emergency Diesel Mission Time and Criteria for Interpreting Lube Oil Trending; dated April 28, 2004
- 003752; Apparent Cause Evaluation: Adverse Trend for Local Leak Rate Test Failures for Flowserve Parallel Double Disk Gate Valves; dated May 1, 2005

- 004267; Apparent Cause Evaluation: Drywell Floor Drain Isolation Valve Failed to Close During Test; dated September 30, 2004
- 007464; Ops Manual Implies That Diesel Generator 13 Required per NUREG 0737; dated April 27, 2004
- 007828; Reactor Building Crane Main Hoist Components Do Not Meet Recommended Factors of Safety; dated April 29, 2004
- 010385; Emergency Diesel Mission Time and Criteria for Interpreting Lube Oil Trending; dated August 1, 2002
- 010841; Degraded Components Found During Underwater Inspections at Intake Structure; dated November 11, 2002
- 012010; Dryer-Separator Storage Pool Leak Detection System Indicates Pool Leak; dated March 11, 2005
- 013061; Reactor Building Crane Main Hoist Components Do Not Meet Recommended Factors of Safety; dated February 10, 2003
- 013780; Several Pipe Support Baseplates; dated February 24, 2003
- 014359; In-Vessel Inspections Found Indications on Jet Pump; dated January 14, 2000
- 018127; IN 2003-20: Derating Whiting Cranes Purchased Before 1980; dated October 29, 2003
- 019081; OE 17615– Excessive Heatup and Cooldown of RPV; dated January 22, 2004
- 019655; IN 2004-05: Spent Fuel Pool Leakage to Onsite Groundwater; dated March 9, 2004
- 021183; OE 17615–Excessive Heatup and Cooldown of RPV; dated February 18, 2004
- 021357; Self Assessment NDE Program; dated January 19, 2005
- 025290; Through Wall Corrosion on Floor of Turbine Lube Oil Storage Tank T-42B; dated September 28, 2005
- 026489; Documentation of in Core Dry Tubes Examinations; dated May 1, 2003
- 026496; License Renewal Aging Effects; dated May 2, 2003
- 026759; Cracks Found in Lug Attachment Welds to Hanger; dated May 2, 2003
- 026787; Cracks Found in Lug Attachment Welds to Hanger; dated May 7, 2003
- 026976; Leakage Identified on LPRM 36-29 During Vessel Hydro; dated May 18, 2003
- 027067; DAEC Indications in the Steam Dryer; dated April 30, 2003
- 027102; Weld Repair for Indication Identified; dated April 29, 2003
- 027162; Core Spray Ultrasonic Testing of Slip Joint Welds; dated May 10, 2003
- 027445; Cracked Tack Weld on Jet Pump; dated May 28, 2003
- 027639; Steam Erosion Found on Piping at Outlet of CV-1243; dated May 16, 2003
- 027905; Piping andInstrumentation Drawing Contains Errors; dated June 19, 2003
- 028223; Dissimilar Metal Welds on Control Rod Drive Scram Header; dated July 15, 2003
- 029744; Inconsistent Bolting on #11 Core Spray Sparger; dated September 25, 2003
- 029949; IN 2003-20: Derating Whiting Cranes Purchased Before 1980; dated October 29, 2003
- 030859; OE-17271 Stress Corrosion Cracking in Closed L; dated November 25, 2003
- 031740; OE 17615–Excessive Heatup and Cooldown of RPV; dated January 22, 2004
- 031967; Identified Hot Spots on Y10-09, Y20-28, and Y-8; dated February 12, 2004
- 032556; IN 2004-05: Spent Fuel Pool Leakage to Onsite Groundwater; dated March 9, 2004
- 033788; Identified a Hot Spot on X60/X80TR-ADISC During Routine Thermography Inspection; dated June 23, 2004
- 034444; Identified Hot Spot in D10 (11 Bay Charger) During Routine Thermography Inspection; dated August 12, 2004

- 034456; Potential Adverse Trend in Deteriorating Control Rod Drive Instrumentation; dated August 13, 2004
- 034891; High Pressure Coolant Injection Isometric Drawing; dated September 20, 2004
- 034915; Support FWH-83; dated September 21, 2004
- 035273; Bolts on EFT Charcoal Filter Access Doors are Deteriorating; dated October 15, 2004
- 036850; Most Limiting Core Component not Used for Determining Crack Growth Mitigation; dated January 27, 2005
- 037309; Instrument Air Filter Analysis From Test 1362 is Out of Spec; dated February 25, 2005
- 037461; Under-Vessel Leakage Inspection Identifies Four Control Rod Drive Flange Leaks; dated March 5, 2005
- 037477; Local Leak Rate Test Failure of MO-2373 and MO-2374 During As Found Test (88 scfh combined); dated March 6, 2005
- 037603; Crack Like Indication Found on Steam Dryer; dated March8, 2005
- 037632; Control Rod Drive Pipe Thru Wall Leakage; dated March 9, 2005
- 037635; Linear Indication was Found in Piping; dated March 9, 2005
- 037675; Dryer-Separator Storage Pool Leak Detection System Indicates Pool Leak; dated March 11, 2005
- 037922; Flow Accelerated Corrosion Damage Identified in Pipe After MO-2008 was Removed; dated March 17, 2005
- 037956; Shroud Inspections Discovered Indications in New Regions; dated March 18, 2005
- 037973; ISI/IWE Indication, ID Pitting Reported on Piping Outboard of -x47 During VT-3; dated March 18, 2005
- 038158; NDE Readings; dated March 24, 2005
- 038523; Shroud Head Bolts 4 and 32 are Degraded; dated April 6, 2005
- 038813; In-Service Inspection Indication, FW-97-1, Erosion on Hinge Pin Plug Hole Threads; dated April 26, 2005
- 038834; Corroded Wires Found in Diesel Fire Pump Control Cabinet; dated April 28, 2005
- 039593; Seat Leakage Identified on Withdrawal Scram Valve; dated June 21, 2005
- 039657; Degraded Wiring of Diesel Fire Pump System; dated June 25, 2005
- 039869; Service Water Piping in Reactor Building Showing Signs of Wear from Metal to Metal Contact; dated July 14, 2005
- 040897; Strap Type Support on Emergency Diesel Generator 11 Ductwork Missing Nut on Anchor Bolt; dated September 21, 2005
- 040967; Through Wall Corrosion on Floor of Turbine Lube Oil Storage Tank T-42B; dated September 27, 2005
- 00694385; IN 2003-20: Derating Whiting Cranes Purchased Before 1980; dated October 29, 2003
- 00708389; Revise UFSAR Fluence Calculation Methodology; dated May 2, 2003
- 00735368; Loose Hangers Found on SW3-JF Piping near the Condensate Pumps; dated July 27, 2004
- 00800385; Roof Leaking onto Turbine Floor Above V-EF-9; dated January 25, 2005
- 00804228; Canal Sample Pump House Has Material Condition Issues; dated February 2, 2005
- 00815635; Several 3" Chunks of Concrete Found Under the Turbine Stop Valves; dated March 5, 2005

- 00838288; Small Holes in Cable Spreading Room Ceiling and Wall, Not Through-Barrier; dated April 26, 2004
- 00859716; NRC Commitment to Retain Capsules Removed from MNGP; dated June 22, 2005
- 0088473-02; Root Cause Evaluation of 13 Diesel Generator Vibration Induced Failure; dated December 16, 2005
- 01007906; Small Areas of Coatings in Containment Did Not Meet Inspection Procedure Acceptance Criteria; dated December 7, 2001
- 03005356; Small Areas of Coatings in Containment Did Not Meet Inspection Procedure Acceptance Criteria, 2003 RFO; dated May 15, 2003
- 03012378; SIL 564, "Verification of SRM, IRM or LPRM Detector Response; dated December 04, 2003
- 98000890; Small Areas of Coatings in Containment Did Not Meet the Acceptance Criteria Specified in Inspection Procedures; dated April 6, 1998

Procedures

- 0135; Pressure-Suppression Chamber Painted Surface Internal Inspection; Revision 8
- 0136; Integrated Primary Containment Leak Rate Test; Revision 13
- 0137; Master Local Leak Rate Test; Revision 25
- 0140; Drywell Interior Surface Inspection; Revision 5
- 0192; Diesel Fuel Quality Check; Revision 18
- 0268; Fire Protection System Flow Test; Revision 15
- 1253; Underground Piping Inspection; Revision 4
- 1350; Underground Storage Tank Liquid Level Correlation; Revision 6
- 1362; Air Quality Test for the Instrument Air System; Revision 7
- 1367; Pressure-Suppression Chamber Below Water Line Painted Surface Internal Inspection; Revision 1
- 1385; Periodic Structural Inspection; Revision 3
- 1396; Equipment/Structures Settling Check; Revision 2
- 1404-01; Emergency Diesel Generator Essential Service Water Heat Exchanger Performance Test; Revision 9
- 1435-1; Underground Storage Tank Quarterly Monitoring; Revision 2
- 1475; Equipment Cycles Surveillance; Revision 0
- 3263; Controlled Specification Data Sheet (Fuel Oil); dated June 20, 1985
- 3802; Visual Inspection of Heat Exchanger Condition; Revision 0
- 8095; Fill Diesel Oil Receiving Tank From Truck; Revision 14
- 8096; Fuel Transfer From Diesel Oil Receiving Tank to Diesel Oil Storage Tank; Revision 5
- 8199; Generator Rotor Lift and Associated Crane Inspection Procedure; Revision 4
- 8236; Application of Nuclear Coatings; Revision 8
- 8280; Torus Painting; Revision 6
- 20-A-10; Southwest Equipment Room V-AC-4 High Temperature; Revision 6
- 20-A-17; Southwest Equipment Room V-AC-5 High Temperature; Revision 6
- 4AWI-06.06.01; Material Handling and Control of Heavy Loads; Revision 11
- 4AWI-07.04.02; Plant Chemistry Program; Revision 3
- 4AWI-09.04.00; Inservice Inspection Licensee Control Program; Revision 3
- 4AWI-09.04.02; System and Component Pressure Testing Program; Revision 11
- 4058-05-PM; "A" RHR Room Air Cooling Unit V-AC-5 Internal Cleaning, External Cleaning and Visual Inspection; Revision 9

- 4125-PM; East Service Water Bay Inspection and Dredging; Revision 9
- 4126-PM; West Service Water Bay Inspection and Dredging; Revision 9
- 4159-PM; Instrument and Service Air Leak Survey; Revision 1
- 4250-01PM; Reactor Building Crane, Bridge Drive System; Revision 20
- 4250-02PM; Reactor Building Crane, Trolley Drive System; Revision 19
- 4260-PM; Refueling Platform Inspection and Lubrication; Revision 22
- 4270-01PM; Turbine Building Crane, Bridge Drive System; Revision 16
- 4270-02PM; Turbine Building Crane, Trolley Drive System; Revision 16
- 4361-PM; Reactor Building Crane Inspection Checklist; Revision 3
- 4858-48-PM; 2R Transformer and Associated Bus PM; Revision 6
- 4858-59-PM; 1R Transformer and Associated Bus Maintenance Procedure; Revision 8
- 4864-PM; Reactor Vessel Head Lifting Device Inspection Procedure; Revision 6
- B.09-15.01; Non-Essential Diesel Generator; Revision 4
- B.8.11-01; Diesel Oil System; Revision 1
- B.9.15; Maintenance Rule Program, System Basis Document Non-Essential Diesel Generator; Revision 2
- C.4-B.09.02.A; Station Blackout Abnormal Procedures; Revision 27
- E.4-01; Backfeed Bus 13 from 13 Diesel Generator; Revision 2
- CD 5.17; Flow Accelerated Corrosion and Service Water Inspection Program Standard; Revision 1
- CD 5.24; Reactor Vessel Integrity Program Standard; Revision 0
- CD 5.25; Generic Letter 89-13 Standard; Revision 0
- CD 5.28; Conduct of Systems Engineering; Revision 2
- ESM-02.02; Design Requirements, Practices and Topics (Mechanical); Revision 11
- EWI-05.02.01; Monticello Maintenance Rule Program Document; Revision 7
- EWI-08.01.01; Boiling Water Reactor Vessel Internals Project (BWRVIP) Administrative Manual; Revision 5
- EWI-08.01.02; BWRVIP Implementation Guidelines; Revision 2
- EWI-08.06.01; Primary Containment Leakage Rate Testing Program; Revision 2
- EWI-08.07.01; Thermal Fatigue Monitoring Program; Revision 0
- EWI-08.19.01; Cable Condition Monitoring Program; Revision 0
- EWI-08.22.01; Generic Letter 89-13; Revision 1
- EWI-08.22.02; Heat Exchanger Condition Assessment Program; Revision 1
- EWI-09.04.00; ASME Section XI Inservice Inspection Program; Revision 0
- FP-IH-EXC-01; Excavation and Trenching Controls; Revision 0
- FP-PE-FAC-01; Flow Accelerated Corrosion Inspection Program; Revision 1
- I.05.30; Sampling Underground Fuel Oil Tanks Monitoring Points; Revision 3
- II.01; Strategic Chemistry Plan; Revision 6
- II.03; Control and Diagnostic Parameters; Revision 2
- II.05; Chemistry Limits and Sampling Frequencies; Revision 15
- MWI-8-M-4.10; Concrete Expansion Bolt Installation; Revision 8
- PEI-02.03.06; Ultrasonic Examination of Bolts and Studs to Appendix VIII; Revision 0
- PEI-02.05.01; Visual Examination; Revision 0
- PEI-02.05.03; Visual Examination of Class MC Components (VT-3); Revision 0
- PEI-02.05.07; Visual Examination (VT-1) of Class MC Components; Revision 0

Procedure Change Requests

- 000511; Flow Accelerated Corrosion Program Fleet Procedure Issue: FP-PE-FAC-01; dated May 20, 2005

Reports

- 04-15FWH to RX; CHECWORKS Wear Rate Analysis: Combined Summary Report; dated October 21, 2005
- 2000U041; Ultrasonic Examination of the Reactor Vessel Nozzle N-2F NV; dated January 12, 2000
- 2001M330; Magnetic Particle Examination: Reactor Pressure Vessel (RPV) Studs; dated November 30, 2001
- 2001U302; Ultrasonic Bolting /Stud Examination: RPV Studs; dated November 9, 2001
- 2001U329; Ultrasonic Examination of the Reactor Vessel Nozzle N-1B NV; dated November 10, 2001
- 2001U331; Ultrasonic Examination of the Reactor Vessel Nozzle N-2A NV; dated November 11, 2001
- 2001U332; Ultrasonic Examination of the Reactor Vessel Nozzle N-2B NV; dated November 11, 2001
- 2001UT025; Ultrasonic Examination of the Reactor Vessel Nozzle N-4C NV; dated March 16, 2005
- 2001UT026; Ultrasonic Examination of the Inner Radius of Reactor Vessel Nozzle N-4C IR; dated March 16, 2005
- 2002-001-5-016; Observation Report: NEC Generic Letter 89-13, Service Water System/Heat Exchanger Program; dated February 28, 2002
- 2002-011; Structural Deficiency Identification and Evaluation Form 4266; dated August 19, 2002
- 2002-024; Structural Deficiency Identification and Evaluation Form 4266; dated August 16, 2002
- 2003-003-5-024; Nuclear Oversight Observation Report: Inservice Inspection Program; dated September 4, 2003
- 2005UT016; Ultrasonic Examination of the Reactor Vessel Nozzle N-2E NV; dated March 19, 2005
- 2005UT042; Ultrasonic Examination of the Inner Radius of Reactor Vessel Nozzle N-1A IR; dated March 27, 2005
- 94-0088; Ultrasonic Examination Report: RPV Studs; dated September 23, 1994
- 98-0009; Ultrasonic Examination Report: RPV Studs; dated March 24, 1998
- 98-016; Structural Deficiency Identification and Evaluation; dated August 3, 1998
- 98-021; Structural Deficiency Identification and Evaluation; dated August 3, 1998
- EHT-416-1; Hardness Test Report, Reactor Vessel Head Stud 47; dated April 16, 1991
- EHT-416-2; Hardness Test Report, Reactor Vessel Head Stud 48; dated April 16, 1991
- EHT-416-3; Hardness Test Report, Reactor Vessel Head Stud 49; dated April 16, 1991
- EHT-416-4; Hardness Test Report, Reactor Vessel Head Stud 50; dated April 16, 1991
- EHT-416-5; Hardness Test Report, Reactor Vessel Head Stud 4-M; dated April 16, 1991
- NP-5911SP; Acceptance Criteria for Structural Evaluation of Erosion-Corrosion Thinning in Carbon Steel Piping; dated July 1988
- SA 021435; Generic Letter 89-13 Program and Ultimate Heat Sink Focused Self-Assessment; dated June 15, 2004
- SASR 89-77; Accumulated Fatigue Usage for Reactor Pressure Vessel; dated December 1989
- UST T-44; Out of Service Inspection Report on Tank T-44; dated May 6, 2003
- XOE02010809; External Operating Experience Notebook: OE15015–Interconnecting Pipe Not in the Flow Accelerated Corrosion Program; dated September 21, 2005

- Containment Inspection Plan (IWE) 1st Interval: September 9, 1998 through May 8, 2008; Revision 1; dated August 2, 2004
- Core Shroud Ultrasonic Examination Final Report; (draft)
- Flow Accelerated Corrosion Master Plan; dated January 8, 2004
- Focused Self-Assessment: Reactor Vessel Thermal Cycles Program; dated March 31, 2004
- Focused Self-Assessment: Flow Accelerated Corrosion and Service Water/MIC Programs, October 27 -30, 2003; dated November 26, 2003
- In Vessel Visual Inspection Final Report, 2005 RFO-22; dated March 28, 2005
- Inservice Inspection Containment Examination Program (Class MC-IWE) Summary Report, Cycle 21 Refueling Outage; dated May 1, 2005
- IWE Outage Summary Report, Refueling Outage 20; dated December 23, 2002
- Maintenance Rule Program: Periodic Assessment Report, June 2003 through May 2004; undated
- Operating Experience Assessment: RICSIL 055: RPV Head Stud Cracking, Revision 1, Supplement 1; dated November 18, 1993
- Out of Service Inspection Report: Diesel Fuel Oil Storage Tank T-44; dated May 6, 2003

<u>Surveillances</u>

- 1253; Underground Piping Inspection; completed October 5, 1987
- 1362; Air Quality Test for the Instrument Air System; completed: February 25, 2005, March 16, 2005, and August 1, 2005
- 1385; Periodic Structural Inspection; completed: August 7, 1998 and August 21, 2002
- 5126; Instrument Air Particulate Sizing; completed: February 25, 2005, March 20, 2005, and August 1, 2005

<u>System and Program Health and Status Reports</u>

- CDR; Main Condenser System; dated January 3, 2006
- CFW; Condensate and Feedwater System; dated January 3, 2006
- CRN; Cranes, Heavy Loads, Rigging System Health Report; dated April 25, 2005
- FAC; Flow Accelerated Corrosion Program; dated January 1, 2006
- FPC; Fuel Pool Cooling and Cleanup System; dated November 3, 2005
- ISI; Inservice Inspection Program; dated November 1, 2005 and December 30, 2005
- MR; Maintenance Rule Program; dated December 1, 2005
- RWCU; Reactor Water Cleanup System; dated January 24, 2006
- TF; Thermal Fatigue Program; dated October 5, 2005

<u>Work Order Written as a Result of the Inspection</u>

- 00003054; Refurbish Support Between SW-10 and SW-11; dated January 26, 2006

<u>Work Orders Reviewed During the Inspection</u>

- 9206184; Clean and Perform Internal Visual Inspection of T-44; dated September 16, 1993
- 9501798; Perform 5 Year Inspection on Turbine Crane; dated November 13, 1995

- 9906790; Inspection of Underground Off Gas Piping Per Procedure 1253; dated September 17, 1999
- 0202752; Clean and Inspect Diesel Fuel Oil Storage Tank; dated May 6, 2003
- 0203364; Replace Lube Oil Cooler Core, 12 Emergency Diesel Generator; dated May 5, 2003
- 0205376; Adjust Reactor Building Crane North Rails; dated November 14, 2002
- 0306092; South Rails on Reactor Building Crane Has Too Big Gap; dated January 24, 2003
- 0306207; Turbine Building Crane: Replace Bolt on Main Hoist Gear Case; dated February 5, 2003
- 0309622; Concrete on NW Corner of High Pressure Coolant Injection Building Chipped Away; dated June 17, 2003
- 0309628; Excessive Corrosion in Sodium Hypochlorite Room; dated June 17, 2003
- 0310892; Perform Electrical and Mechanical Preventive Maintenance on Reactor Building Crane Bridge, Trolley, Main and Auxiliary Hoist; dated October 1, 2003
- 0311446; Inspect and Repair Sheaves for Reactor Building Crane Auxiliary Hoist; dated November 26, 2003
- 0401172; Hydrolaze Division II FSW Piping; dated March 22, 2005
- 0402742; Tighten Hangers on Service Water System; dated August 3, 2004
- 0508437; Replace Missing Nuts on Fuel Pool Cooling Supports; dated October 31, 2005

Work Requests

- 92-06184; Inspection and Cleaning of Diesel Fuel Oil Storage Tank; dated February 11, 1993
- 92-06189; Flushing of 11 Emergency Diesel Generator Day Tank; dated March 8, 1993
- 92-06192; Flushing of 12 Emergency Diesel Generator Day Tank; dated February 11, 1993

Monticello Nuclear Generating Plant

Scoping, Screening and Aging Management
License Renewal Inspection
Exit Meeting
February 22, 2006

Agenda

- Introductions
- License Renewal Process
- NRC Inspection Results
- Applicant Comments
- Public Questions

License Renewal Process

- Federal regulations (10 CFR Part 54) allow for renewal of operating licenses for an additional 20 year period
- Monticello submitted its application for a renewed license on March 24, 2005

License Renewal Process, cont.

- Current license granted for 40 year period; it will expire on September 8, 2010
- Operating license would be extended to 2030
- Day-to-day operations under 10 CFR Part 50 not changed

License Renewal Process, cont.

License Renewal Process includes

- Technical Review by Office of Nuclear Reactor Regulation (NRR)
  - Onsite Scoping Audit
  - Onsite Aging Management Audit
  - Culminates in Safety Evaluation Report

License Renewal Process, cont

- Finally, process includes onsite inspection by regional office
  - Focuses on implementation and commitment management
  - Emphasizes in-plant walk downs
  - Culminates in an inspection report

Inspection

- Onsite inspection performed in accordance with NRC Inspection Procedure 71002
- Inspected scoping, screening and aging management programs

Inspection, cont.

- Inspection occurred from January 23 to February 10, 2005
- Consisted of two weeks onsite
- Inspection team consisted of five experienced inspectors plus license renewal project manager

Inspection, cont.

- Team reviewed electrical, mechanical and structural systems, structures and components
- Normally inaccessible areas inside containment will be examined by routine inspections during next refueling outage

Scoping and Screening

- Reviewed 12 systems to verify appropriateness of scoping and screening efforts
  - Supplemented NRR's review
  - Emphasized physical walk downs of the plant
  - Concentrated on systems, structures and components not included within the applicant's license renewal scope

Scoping and Screening, cont.

- Systems reviewed included:
  - Condensate and Feedwater
  - Diesel Generator Starting Air
  - Drywell Atmosphere Cooling
  - Fuel Pool Cooling and Cleanup
  - Heating and Ventilation
  - Instrument Air

Scoping and Screening, cont.

- Conclusion:
  - Systems generally appeared appropriately scoped and screened
  - Some minor inconsistencies identified
  - In some cases, the changes required a revision to the application; this will be part of the annual application update
  - Scoping and screening acceptable for license renewal

Aging Management Programs

- Reviewed 33 aging management programs and 2 time limited aging analyses (TLAA) programs
  - Followed up on NRR review efforts
  - Performed plant walk downs
  - Reviewed existing plant documentation

Aging Management Programs, cont

- Reviewed operational experience information
- Reviewed corrective actions to current plant issues and
- Reviewed proposed enhancements and commitments
- Verified that TLAAs were appropriate

Aging Management Programs, cont

- The programs reviewed included:
  - ASME Section XI (2 programs)
  - Buried Piping & Tanks Inspection
  - Bus Duct Inspection Program
  - BWR Nozzles and Penetrations (3 Programs)
  - BWR Stress Corrosion Cracking & Neutron Irradiation Embrittlement (2 Programs)
  - BWR Vessel Internals and Attachments (2 Programs)

Aging Management Programs, cont

- Programs reviewed, cont.
  - Compressed Air Monitoring
  - Open & Closed Cycle Cooling Water Systems (2 Programs)
  - Plant Chemistry Program
  - Primary Containment Inspection & Leak Rate Testing (2 Programs)
  - Protective Coating Program

Aging Management Programs, cont

- Programs reviewed, cont.
  - Non-EQ Cables (3 Programs)
  - Fire Protection & Fire Water (2 Programs)
  - Flow-Accelerated Corrosion
  - Fuel Oil Chemistry
  - Inspection of Overhead Cranes
  - One-Time Inspection & Selective Leaching (2 Programs)

Aging Management Programs, cont

- Programs reviewed, cont.
  - Reactor Head Closure Studs & Bolting Integrity (2 Programs)
  - Reactor Vessel Surveillance
  - Structures & System Condition Monitoring (2 Programs)
- Time Limited Aging Analyses
  - Environmental Qualification
  - Fatigue Monitoring

Aging Management Programs, cont

- Conclusion
  - Existing aging management programs generally implemented as described in the application
  - Enhancements and exceptions appeared acceptable and were captured in commitment tracking database

Aging Management Programs, cont

- Conclusion, cont.
  - Some minor inconsistencies identified which either required revision to the application or documentation in the corrective action program
  - Application revisions will be part of the annual application update
  - Aging Management Programs should be adequate for period of extended operation

Inspection Conclusions

- Monticello scoping, screening and aging management programs sufficient for extended operation
- Region III does not see any inspection impediments to renewing the operating license

LIST OF ACRONYMS USED

ADAMS	Agency Wide Access Management System
AMP	Aging Management Program
AMSAC	ATWS Mitigation System Actuating Circuitry
AR	Action Request
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel Internals Project
CASS	Cast Austenitic Stainless Steel
CCCW	Closed Cycle Cooling Water
CFR	Code of Federal Regulations
CFW	Condensate and Feedwater Systems
CRDRL	Control Rod Drive Return Line
EF	Degree Fahrenheit
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling Systems
EQ	Environmental Qualification
EVT	Enhanced Visual (Test) Examination
FAC	Flow Accelerated Corrosion
FP	Fire Protection
GALL	Generic Aging Lessons Learned
GE	General Electric
HTV	Heating and Ventilation
IASCC	Irradiation Assisted Stress Corrosion Cracking
ID	Inside Diameter
ISG	Interim Staff Guidance
ISI	Inservice Inspection
ISP	Integrated Surveillance Program
ksi	Kilo (1000) Pounds per Square Inch
kV	kiloVolt (1000 Volts)
LER	Licensee Event Report
LR	License Renewal
LRA	License Renewal Application
MNGP	Monticello Nuclear Generating Plant
NMC	Nuclear Management Company, LLC
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OCCW	Open Cycle Cooling Water
PARS	Publically Available Records System
PCT	Peak Clad Temperature
PM	Preventive Maintenance
RG	Regulatory Guide
RSW	Residual Heat Removal Service Water
SIL	Service Information Letter
SSC	System, Structure, or Component
SSW	Service and Seal Water
TLAA	Time Limited Aging Analyses

USAR	Updated Safety Analysis Report
VT	Visual (Testing) Examination