November 27, 2001

Mr. Kurt M. Haas
General Manager
Big Rock Point Nuclear Plant
Consumers Energy Company
10269 US 31 North
Charlevoix, MI 49720

SUBJECT: BIG ROCK POINT INSPECTION REPORT 05000155/2001-006(DNMS)

Dear Mr. Haas:

On November 8, 2001, the NRC completed an inspection at the Big Rock Point Nuclear Plant Restoration Project. The focus of the inspection activities was on facilities management and control, decommissioning support activities, spent fuel safety, and radiological safety. Activities included a meeting with the Big Rock Point Citizen's Advisory Board on November 6, 2001. The enclosed report presents the results of these inspection activities.

Overall, reactor decommissioning activities were being performed satisfactorily. No violations of NRC requirements were identified.

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

We will gladly discuss any questions you may have regarding this inspection.

Sincerely,

/RA/

Bruce L. Jorgensen, Chief
Decommissioning Branch

Docket No. 05000155
License No. DPR-6

Enclosures: 1) Inspection Report 05000155/2001-006(DNMS)
2) Handouts from November 6, 2001 Meeting

See Attached Distribution

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D. Minnaar, Michigan Department of
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Chief, Nuclear Facilities Unit, Michigan
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	05000155
License No:	DPR-06
Report No:	05000155/2001-006(DNMS)
Licensee:	Consumers Energy Company
Facility:	Big Rock Point Nuclear Plant
Location:	10269 U.S. 31 North Charlevoix, MI 49720
Dates:	October 15-November 8, 2001
Inspectors:	William Snell, Health Physics Manager Ross Landsman, Project Engineer Dave Wrona, Project Manager, NRR
Approved By:	Bruce L. Jorgensen, Chief Decommissioning Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Big Rock Point Restoration Project
NRC Inspection Report 05000155/2001-006(DNMS)

This routine decommissioning inspection covered facilities management and control, decommissioning support activities, spent fuel safety and radiological safety. Overall, major decommissioning activities were properly monitored and controlled.

Facilities Management and Control

- The Big Rock Point Restoration Safety Review Committee (RSRC), an outside peer review group, was observed to have a questioning attitude as well as a different perspective on issues, which appeared to be beneficial to the licensee's program. (Section 1.1)

- The licensee's safety review program conformed to the requirements contained in 10 CFR 50.59. Several safety reviews evaluated by the inspectors were performed by personnel trained on the latest version of Procedure D1.11 and appropriately determined that NRC approval was not required. (Section 1.2)

Decommissioning Support Activities

- Adequate actions were taken by the licensee to ensure that systems important to safe storage of spent were protected against extreme cold weather. (Section 2.1)

Spent Fuel Safety

- The individual members, connections, and anchorages for the new 125 ton containment building crane met the applicable codes and standards. The use of the new crane should result in no undue risk to the movement of fuel. (Section 3.1)

- Buried utilities along the dry cask haul route to the Independent Spent Fuel Storage Installation (ISFSI) had been evaluated to ensure their integrity or appropriate bridging provided. The subgrade, subbase, and pavement had also been evaluated to provide a safe pathway. The haul route appeared satisfactory. (Section 3.2)

- All the support structures under the containment building access bridge were appropriately evaluated to safely transport a loaded cask out of the containment building. (Section 3.3)

- The ISFSI pad conformed to the Certificate of Compliance requirements. (Section 3.4)

Radiological Safety

- The licensee's program for the calibration of radiation survey instruments was reviewed and determined to be functioning acceptably. A problem was identified with the printing out and on-screen view of the due dates for calibration of radiation survey instruments, but it did not appear to have ever resulted in an instrument failing to be calibrated when required. (Section 4.1)

Report Details[1]

1.0 Facilities Management and Control

1.1 Restoration Safety Review Committee Meeting (40801)

a. Inspection Scope

The inspectors observed meetings by the Big Rock Point Restoration Safety Review Committee.

b. Observations and Findings

Administrative Procedure D1.8, *Big Rock Point Restoration Safety Review Committee (RSRC),* Revision 0, establishes the responsibilities for the RSRC. The purpose of the RSRC is to assist and advise licensee management on issues important to the safety and economic success of the restoration project. RSRC members meet at least twice per year and are provided site documents for review prior to meetings.

The RSRC, composed of five individuals from outside Consumers Energy Company, met with licensee personnel on November 6-7, 2001. The committee and plant staff and management were observed candidly discussing all aspects of the restoration program. The dialogue was open and with a clear intent to improve the program. Having a peer review group from outside the company appeared to be a benefit to the licensee’s decommissioning program.

c. Conclusions

The Big Rock Point Restoration Safety Review Committee (RSRC), an outside peer review group, was observed to have a questioning attitude as well as a different perspective on issues, which appeared to be beneficial to the licensee’s program.

1.2 Safety Reviews, Design Changes, and Modifications (37801)

a. Inspection Scope

The inspectors reviewed the licensee’s safety review process and procedures to determine whether the program conformed to 10 CFR 50.59. The following items were reviewed:

- Procedure D1.11, "10 CFR 50.59 and 10 CFR 50.82 Evaluations," Revision 3
- Procedure D3.1.1.1, "Facility Changes," Revision 4
- Procedure D3.1.1.3, "Setpoint Changes," Revision 2
- Procedure D3.1.1.4, "Jumpers, Links, and Bypasses/Temporary Modifications," Revision 2
- Procedure D3.1.1.8, "Minor Alterations," Revision 5
- Form BRP124, "Big Rock Point 10 CFR 50.59 and 10 CFR 50.82 (PSDAR) Evaluation"

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[1]NOTE: A list of acronyms used in the report is included at the end of the Report Details.

b. Observations and Findings

Procedure D1.11 provided the licensee's process for meeting the intent of 10 CFR 50.59. The inspectors verified that the requirements in Procedure D1.11 conformed to 10 CFR 50.59. For example, the criteria specified in 10 CFR 50.59 for determining whether NRC approval was required for a change, test, or experiment were adequately addressed. The inspectors noted a minor difference in the wording of the criteria used to evaluate if NRC approval was required between Procedure D1.11 and Form BRP124. The licensee had previously identified the inconsistency and initiated QR Log #387-01 to correct Procedure D1.11.

Procedures D3.1.1.1, D3.1.1.3, D3.1.1.4, and D3.1.1.8 appropriately instructed personnel to consider performing safety evaluations in accordance with the instructions contained in Procedure D1.11 for items that could be changes, tests, or experiments as defined in 10 CFR 50.59.

The inspectors reviewed the following safety evaluations prepared for various activities performed over the past year:

- Form BRP124 for Minor Alteration MA-01-0022, "Core Sampling Beneath Containment," Revision 0
- Form BRP124 for Field Change FC-0708, "Remove Structure for Reactor Vessel (RV) Removal," Revision 1
- Form BRP124 for Jumper, Link and Bypass/Temporary Modification JLB-01-0011, “Spent Fuel Pool Debris Barrier Installation and Removal”

The safety evaluations were prepared and reviewed by personnel who were trained on the latest revision to Procedure D1.11. The inspectors determined that the licensee's conclusions that NRC approval was not required were appropriate.

c. Conclusions

The licensee's safety review program conformed to the requirements contained in 10 CFR 50.59. Several safety reviews evaluated by the inspectors were performed by personnel trained on the latest version of Procedure D1.11 and appropriately determined that NRC approval was not required.

2.0 Decommissioning Support Activities

2.1 Cold Weather Preparations (71714)

a. Inspection Scope

The inspector evaluated the licensee’s actions to effectively protect safety-related systems against extreme cold weather.

b. Observations and Findings

The inspector reviewed the implementation of Procedure No. O-VAS-5B, *Cold Weather Checklist*, Revision 2. The purpose of the procedure was to align ventilation systems and various plant equipment for cold weather operations. The checklist was initiated on

September 9, 2001, and was signed off as completed on November 1, 2001. A review of the Checklist determined that all steps had been appropriately signed off and dated as required.

c. Conclusion

Adequate actions were taken by the licensee to ensure that systems important to safe storage of spent were protected against extreme cold weather.

3.0 Spent Fuel Safety

3.1 Containment Building Crane Upgrade (60853)

a. Inspection Scope

This portion of the inspection evaluated whether the methodology used for analysis and design of the components associated with the new containment crane encompassed NRC requirements (codes and standards).

b. Observations

Big Rock Point had a 75 ton containment building crane that was used for construction and operating loads. In 1999, a contract was issued for the replacement of the 75 ton crane with a single failure proof 125 ton crane in order to load the dry fuel storage casks. Ederer was selected as the single failure proof trolley/hoist system vendor. Ederer's single failure proof design is documented in topical report EDR-1(P)-A, Revision 3, dated October 8, 1982, which was reviewed and approved by the NRC in August of 1983.

In order to support the new trolley, the project is also installing a replacement gantry structure as part of the crane upgrade. During analysis, some of the seismic load cases indicated up-lift for the gantry trucks at the 632'-6" elevation. A modification to prevent the trucks from leaving the rails was determined to be impractical. This unrestrained up-lift resulted in revising the maximum critical load of the crane to 105 tons, to remove the up-lift for the gantry trucks. This was consistent with the maximum cask weight needed for removal of spent fuel. It should be noted that the seismic analysis also included consideration of the pendulum effect for the lifted load.

In addition to the new trolley and gantry structure, an evaluation of the in-place building components such as the rails, rail clips, runway girders and steel supports, and existing supporting concrete was performed. The rail clips and anchorage required modification. The existing concrete structures were found to be adequate for the additional crane loads with the following limitations:

- All piping connected to the steam drum is to be removed before crane use.

- At elevation 660'-6", the live load including any attachments, is limited to 20 pounds per square foot (psf) during crane operation.

- At the slab at elevation 632'-6", between the reactor cavity wall and the north wall of the steam drum enclosure, the attachment load is limited to 20 psf and the live load is limited to 20 psf during crane operation.

New crane load testing requires that static and dynamic load tests must be performed in all positions generating maximum strain in the bridge and trolley structures. Because of movement limitations, testing will only be done over the western 50 feet of the bridge runway so as not to transport the test load over the spent fuel pool. All other testing will provide reasonable assurance of crane operation.

During the 125 percent load test, the test weights began to tilt. This was of concern because the test weights consisted of 20 ton steel slabs resting on top of each other, such that one or more test weights could slide off each other. The test was stopped until the crane manufacturer could design another rigging method to prevent tipping. The test was successfully completed October 25, 2001.

Crane operator training was done in accordance with ANSI B30.2 guidelines. A defined safe load path and procedural controls provide further assurance of safe handling of the casks.

c. Conclusions

The individual members, connections, and anchorages for the new 125 ton containment building crane met the applicable codes and standards. The use of the new crane should result in no undue risk to the movement of fuel.

3.2 Transportation Haul Route (60853)

a. Scope

The inspection evaluated whether the licensee has ensured the safe and proper route for the loaded casks between the containment and the Independent Spent Fuel Storage Installation (ISFSI) pad.

b. Observations and Findings

The haul route was designed using the AASHTO Flexible Pavement Design Method to determine the required roadway base and pavement thickness. The roadway was designed to ensure that the haul trailer does not bottom out, and it was made as straight as possible to minimize turning with a loaded cask.

Analyses have been performed using ground penetrating radar geophysical surveys to verify the integrity of the former railroad roadbed. The readings didn’t indicate any large voids. Subsequent proof rolling and soil testing demonstrated that identified questionable areas were acceptable and the subgrade was compacted to 95 percent of Standard Proctor Density.

The capacities of a storm drain and fire loop piping under the roadway to withstand the design loads were evaluated. The fire loops were found to be adequate to support the loading. Bridging was used over the 18-inch corrugated storm drain to prevent its possible failure.

Buried electrical cable and conduit were at a sufficient depth to provide adequate protection from the haul trailer. Three electrical pull boxes were bridged to prevent their possible failure.

To obtain the least roadway slope, the haul route had to be raised approximately three feet at the ISFSI pad and containment loading dock. The fill was placed and compacted to 95 percent of Standard Proctor Density. Twelve inches of roadway base material were placed and compacted to 98 percent of Standard Proctor Density. The haul road surface consists of three inches of asphalt.

c. Conclusions

Buried utilities along the dry cask haul route to the ISFSI had been evaluated to ensure their integrity or appropriate bridging had been provided. The subgrade, subbase, and pavement had also been evaluated to provide a safe pathway. The haul route appeared satisfactory.

3.3 Air Pallet Bridge (60853)

a. Scope

The inspection evaluated the modifications and analyses performed to ensure the containment building structures could safely support the cask as it is moved onto the loading dock.

b. Observations and Findings

The cask exit path from inside the containment sphere is through an opening cut in the sphere, then over the Post Incident Room. Because air pallets are to be used to move the casks, a level bridge had to be constructed over the various obstacles. The major components of the support structure for the bridge were support posts, load spreaders, and underground wall bracing in the Post Incident Room.

This route will also be used for removal of the steam drum, reactor vessel, and the Bigge Crane erection crane; therefore, support posts were also added for these loads. In addition, the air pallets could fail at any time, so provisions on the bridge were also made to jack the cask up at any point along the bridge span. This resulted in inclusion of bridge main I-beam stiffeners.

c. Conclusions

All the support structures under the containment building access bridge were appropriately evaluated to safely transport a loaded cask out of the containment building.

3.4 ISFSI Pad (60853)

a. Scope

The inspection evaluated whether the cask storage pad had been adequately designed and constructed to support the static load of the casks.

b. Observations and Findings

The pad was designed to meet the requirements of the Certificate of Compliance (C of C). It was sized to accommodate seven canisters, with one greater than class C container housed inside a storage cask. The pad was constructed approximately 300 yards south of the existing Radwaste Building near an abandoned railroad line.

The site required excavation into a small sand dune which resulted in east and south slopes adjacent to the facility. Slope stability analyses were performed for the slopes on the east and south sides of the pad. The analyses concluded that design slopes with a 2.8 horizontal to 1 vertical slope would provide an adequate margin of safety. Based on the pad elevation and a hydrological study, flood waters are not considered a problem.

Finite element models of the concrete pad and soil beneath it were analyzed to determine moments, spheres, settlements and the subgrade bearing pressure for the loadings imposed. The allowable bearing capacity, settlement, and other geotechnical parameters of the soil beneath were obtained from detailed geotechnical investigations. These included seismic cross hole slots. The seismic input was in accordance with Big Rock Point's Regulatory Guide 1.60 response spectra with a peak zero period horizontal ground acceleration of 0.12 grams. The subgrade soil properties met all the C of C parameters.

The 24-36 inch pad was poured on June 30, 2001. All the test cylinder breaks were within the C of C specification requirements. The requirements of ACI 318 were invoked through specification provisions by the construction contractor.

An alternate lightning protection system has been employed than the one identified in the C of C. Instead of grounding the casks, five lightning protection masts have been designed to use a 100 foot radius zone-of-protection rolling sphere model. Each mast is 52 feet in height, including a four-foot blunt-tipped air terminal attached to the top. This design ensures that the dimensional sphere of the rolling ball model does not intercept the casks on the pad. The protection system conforms to NFPA-780 Lightning Protection Codes 96 and 96A.

c. Conclusions

The ISFSI pad conformed to the Certificate of Compliance requirements.

4.0 Radiological Safety

4.1 Occupational Radiation Exposure (83750)

a. Inspection Scope

The inspectors reviewed several areas of the licensee's radiological controls program, including the calibration of radiation survey instruments and various Radiation Work Permits for high dose work.

b. Observations and Findings

The inspectors reviewed the licensee's program for tracking and calibration of radiation survey instruments. The licensee had a comprehensive computerized instrument calibration data base that maintained a record of the past performance of all radiation survey instruments and the dates when instrument calibrations were due. An inspection of the list, as well as a visual inspection of survey instruments in use throughout the site, indicated that all instruments were being maintained calibrated within the required due dates. However, the inspector did note that on the computer printout, as well as when viewed visually on the computer screen, the "update" and "due date" columns in the calibration data base were missing the left most digit. This resulted in the "1" being deleted from the months of October, November and December, so they were printed or viewed as "0", "1" and "2", instead of "10", "11" and "12", respectively. This could have resulted in a failure to perform a calibration in a timely manner; however, an initial review of the data base did not identify any discrepancies. The licensee wrote a Condition Report (C-BRP-01-0312) on the issue.

The inspectors also toured the licensee's calibration facility, and discussed the procedures used to calibrate the various survey instruments. The technician in charge of conducting the calibrations appeared to very knowledgeable of and technically capable to conduct the calibrations.

The inspectors reviewed five Radiation Work Permits (RWPs) that dealt with high dose work. The review was to ensure the scope of the RWP and the working requirements specified were appropriate for the work being performed and consistent with ALARA (as-low-as-reasonable-achievable) principles. The RWPs reviewed were: RWP B010001, *Routine Radiation Protection & Chemistry Activities*, RWP B010003, *Maintenance General Work Activities*, RWP B010283, *Locked High Radiation Area Entries*, RWP B013039, *Reactor Vessel Preparation-Reactor Deck*, and, RWP B013044, *Reactor Vessel Control Rod Drive Room*. The RWPs were appropriate for the work being conducted. In addition, in those situations where unexpected problems had arisen during work activities, the licensee had conducted in-progress ALARA reviews and modified the RWPs accordingly.

c. Conclusions

The licensee's program for the calibration of radiation survey instruments was reviewed and determined to be functioning acceptably. A problem was identified with the printing out and on-screen view of the due dates for calibration of radiation survey instruments, but it did not appear to have resulted in an instrument failing to be calibrated when required.

5.0 Citizens Advisory Board Meeting

On November 6, 2001, NRC staff and management representatives attended a meeting of the Big Rock Point Citizens Advisory Board (CAB). The NRC had sought the opportunity to inform the Board of NRC activities regarding the Big Rock Point decommissioning project, and to solicit stakeholder input into the development of the Regional Master Inspection Plan (MIP) for Big Rock Point. Each year the Region develops a MIP to facilitate the efficient allocation of inspection resources and specify the inspection effort planned for each inspection procedure. During the meeting the NRC described the Agency's decommissioning inspection program, discussed the licensee's performance over the previous 12 months, answered questions from the CAB members, and provided an opportunity for any input regarding the development of the MIP for CY2002 (see Enclosure 2).

6.0 Exit Meeting

The inspectors presented initial inspection results to members of licensee management at the conclusion of the inspection on November 8, 2001. The licensee acknowledged the findings presented. The licensee did not identify any documents or processes reviewed by the inspectors as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

K. Haas, Plant General Manager
M. Bourassa, Licensing Supervisor
K. Pallagi, Radiation Protection and Environmental Services Manager
G. Petitjean, Licensing Supervisor
G. Withrow, Engineering, Operations & Licensing Manager

INSPECTION PROCEDURES USED

IP 37801 Safety Reviews, Design Changes, and Modifications
IP 40801 Self Assessment, Auditing, and Corrective Action
IP 60853 On-Site Fabrication and Construction of an ISFSI
IP 71714 Cold Weather Preparations
IP 83750 Occupational Radiation Exposure

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
CAB	Citizens Advisory Board
C of C	Certificate of Compliance
CR	Condition Report
CY	Calendar Year
ISFSI	Independent Spent Fuel Storage Installation
MIP	Master Inspection Plan
NRC	Nuclear Regulatory Commission
RP	Radiation Protection Technicians
RSRC	Restoration Safety Review Committee

PARTIAL LIST OF LICENSEE DOCUMENTS REVIEWED

Licensee documents reviewed and utilized during the course of this inspection are specifically identified in the "Report Details" above.

BIG ROCK POINT
DECOMMISSIONING PERFORMANCE

October 1, 2000 - September 30, 2001

Decommissioning Support Activities

- Maintenance of material condition of the plant was good.

- Defueled Technical Specifications were being followed.

- Spent fuel were protected against extreme cold weather.

- Physical security was being implemented in accordance with the Security Plan.

BIG ROCK POINT
DECOMMISSIONING PERFORMANCE

October 1, 2000 - September 30, 2001

Spent Fuel Safety

- The Spent Fuel Pool monitors (temperature, level, radiation) were functioning properly.

- The licensee was making acceptable progress towards dry fuel storage.

BIG ROCK POINT
DECOMMISSIONING PERFORMANCE

October 1, 2000 - September 30, 2001

Radiological Safety

- The licensee has a sound radiation protection program.

- The licensee's solid radwaste and transportation programs completed all shipments in compliance with Department of Transportation and NRC regulations.

- Environmental sampling initiatives related to site characterization have been well performed.

BIG ROCK POINT
DECOMMISSIONING PERFORMANCE

October 1, 2000 - September 30, 2001

Facility Management and Control

- The Big Rock Point staff and management were doing a good job in implementing their Decommissioning Plan in a safe and effective manner.

- The licensee's Corrective Action process was well implemented.

- The Nuclear Performance Assessment Department (NPAD) was effective in identifying, resolving, and preventing issues that would degrade safety or the quality of decommissioning.

- The licensee adequately demonstrated the implementation of their Defueled Emergency Plan during a practice drill.

NRC Region III

Bruce Jorgensen, Chief, Decommissioning Branch	630-829-9615
Bill Snell Health Physics Manager	630-829-9871
Pam Alloway-Mueller Public Affairs	630-829-9662
Jan Strasma Public Affairs	630-829-9663

U.S. NRC
Region III
801 Warrenville Road
Lisle, IL 60532-4351

NRC Headquarters

Dave Wrona, Project Manager Big Rock Point	301-415-1924

NRC Web Site

http://www.nrc.gov

DECOMMISSIONING POWER REACTOR INSPECTION PROGRAM

PURPOSE

To establish the inspection policy and guidance for decommissioning power reactors.

OBJECTIVES

To obtain information through direct observation and verification of licensee activities to determine whether the power reactor is being decommissioned safely, that spent fuel is safely stored onsite or transferred to another licensed location, and that site operations and license termination activities are in conformance with applicable regulatory requirements, licensee commitments, and management controls.

To ensure that the licensee's systems and techniques for decommissioning and license termination activities are adequate and in accordance with regulatory requirements. These systems include, in part, management and organization effectiveness; self-assessment, auditing, and corrective actions; design control; maintenance and surveillance; radiation protection; radioactivity measurements; and, effluent controls.

To identify declining trends in performance and perform inspections to verify that the licensee has resolved the issue(s) before performance declines below an acceptable level.

To provide for effective allocation of resources for the inspection of power reactors following permanent cessation of operation.

INSPECTION PROCEDURES

The fundamental objectives of the inspection program are implemented through the use of inspection procedures. These procedures are divided into functional area assessments to inspect licensee performance, identify performance trends, preclude problems, identify weaknesses, and foster corrective actions to contribute to public health and safety and the protection of the environment. The inspection program provides appropriate latitude for NRC management to administer, plan, and implement site-specific master inspection plans commensurate, in part, with licensee performance, site activities, and safety. The inspection effort associated with the implementation of the inspection procedures is, in part, dependent on the decommissioning activities being planned or performed at the facility.

MASTER INSPECTION PLANS

A site-specific master inspection plan (MIP) is developed for reactor power facilities undergoing decommissioning. The intent of the site-specific master inspection plans is to facilitate the efficient allocation of inspection resources, list the inspection effort planned for each inspection procedure, identify the lead inspector and dates of the inspections, and be specific for each facility.

The following are factors that are considered when developing and implementing a site-specific master inspection plan:

> Design. Some power reactors will have unique designs, configurations, and environmental considerations that would cause an inspection effort to focus on specific areas of potential concern. Similarly, technologically advanced contamination removal

methods, dismantlement techniques, or transportation packaging may call for enhanced NRC monitoring.

Plant Status. Plant status will vary depending on the phase of decommissioning. Master inspection plans will take into account the licensee's spent fuel storage location and transfer plans; criticality and decay heat removal considerations; decommissioning fund status; and, planned facility and environmental changes.

Licensee Performance, Management, and Decommissioning Scheduling. The site-specific master inspection plans will be based on licensee performance, staffing plans, effectiveness of management oversight and contractor control, and the timing and scheduling of significant decommissioning activities. Other elements such as the loss of licensee technical expertise and nuclear experience may also factor into the development of a site-specific inspection plan.

The staff will periodically review the site-specific master inspection plans and adjust the plans to reflect inspection findings or changes in plant status and decommissioning activities. During these plan reviews, regional management is involved in the assessment of decommissioning licensee performance and uses these insights as one of the many possible justifications to change the site-specific master inspection plan (i.e., increased or decreased inspection effort, schedule changes, or deletions). This review of licensee performance is focused on the four major inspection areas of the decommissioning power reactor inspection program: facility management and control; decommissioning support activities; spent fuel safety; and, radiological safety. NRC management may also choose to periodically visit the decommissioning site to meet with licensee representatives to evaluate the current status of decommissioning activities and to gain licensee management insights and perspectives. In addition, State representation and other interested parties may be invited to attend these visits to obtain input from all stakeholders.

SUMMARY

In summary, the inspection program for power reactor facilities that have permanently shut down emphasizes a balanced look at a cross section of licensee activities important to the conduct of safe decommissioning. Licensee decommissioning programs and procedures are assessed to ensure that they afford a comparable level of quality, rigor, and effectiveness as those in existence during reactor power operations. The inspection program also provides Regional Administrators flexibility in the application of inspection resources to deal with issues and problems at specific plants.

CORE INSPECTION PROCEDURES
FOR DECOMMISSIONING POWER REACTORS

Facility Management and Control

36801 Organization, Management & Cost Controls
37801 Safety Reviews, Design Changes, and Modifications
40801 Self-Assessment, Auditing, and Corrective Action
71801 Decommissioning Performance and Status Review

Decommissioning Support Activities

62801 Maintenance and Surveillance
71714 Cold Weather Preparations
81700 Physical Security Assessment
81001 Independent Spent Fuel Storage Installations (ISFSI)

Spent Fuel Safety

60801 Spent Fuel Pool Safety
60705 Preparation for Reactor Fuel Handling
60710 Fuel Handling Activities
60851 Design Control of ISFSI Components
60852 ISFSI Component Fabrication by Outside Fabricators
60853 Onsite Fabrication of Components and Construction of an ISFSI
60854 Preoperational Testing of an ISFSI
60855 Operation of an ISFSI
60856 Review of Licensee 72.212(b) Evaluations

Radiological Safety

83750 Occupational Radiation Exposure
83801 Inspection of Final Surveys
84750 Radioactive Waste Treatment, and Effluent & Environmental Monitoring
86750 Solid Radioactive Waste Management & Transportation of Radioactive Material