June 20, 2000

Ms. Corinne Carey
2213 Riverside Drive, NE
Grand Rapids, MI 49505

Dear Ms. Carey:

Thank you for the opportunity to respond to the questions included in your letter dated March 24, 2000. Your questions followed the NRC public meeting conducted on March 23, 2000, in Bridgman, Michigan, out of concern for public health and environmental safety issues. The NRC has as its primary mission the protection of public health and safety, as it relates to the regulated activities, including activities at D.C. Cook Nuclear Power Plant. Your questions were appreciated.

We have answered each of your questions using a format similar to your letter. The enclosed question and answer (Q&A) format includes a restatement of your question, followed by the answer. Some of the answers have supporting documentation and reference specific enclosures to this letter. Each enclosure is clearly marked at the top.

Thank you for your interest in the public and environmental safety issues associated with the D.C. Cook Nuclear Power Plant. Again we thank you for the opportunity to respond to your questions.

Sincerely,

/RA by Steven A. Reynolds Acting For/

John A. Grobe, Director
Division of Reactor Safety

Enclosures:
1. Questions and Answers
2. D.C. Cook Annual Radioactive Effluent Release Report
3. D.C. Cook Off-Site Dose Calculation Manual
4. Table of Half-Lives
5. NRC Inspection Report Nos. 50-315/97014(DRS); 50-316/97014(DRS)
6. NRC Inspection Report Nos. 50-315/99005(DRS); 50-316/99005(DRS)
7. NUREG 1600
8. Notice of Violation and Proposed Imposition of Civil Penalty dated 10/13/98

cc w/encl 1: A. Vegel, DRP

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cc w/encl 1: A. Vegel, DRP

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Enclosure 1

Questions:

1. **Did you, and how, consider and provide data on full air and water impacts of restarting Cook?**

   The U.S. Nuclear Regulatory Commission (NRC) is an independent agency established by the U.S. Congress under the Energy Reorganization Act of 1974 to ensure adequate protection of the public health and safety, the common defense and security, and the environment in the use of nuclear materials in the United States. Ensuring adequate protection of the public health and safety and the environment are two major cornerstones of the agency mission. To meet these objectives the NRC regulates the use of nuclear power through specific regulations and license requirements. Compliance with these requirements is verified through the NRC independent inspection program. The NRC conducts specific independent inspection activities related to air and water effluents from nuclear power plants to ensure adequate protection of the public health and safety and the environment. In answering the remaining questions specific documents will be referenced and are attached.

   In compliance with specific license commitments and NRC regulatory requirements, D.C. Cook Nuclear Plant is required to provide a detailed Annual Radioactive Effluent Release report. Enclosed is a copy of the 1996 Annual Radioactive Effluent Release Report (Enclosure 2) that will be referenced in answering your questions. This report was chosen because it was the last 12 month period of full power operation of the D.C. Cook Nuclear Plant and will more closely represent effluent measurements in the environment surrounding the plant during operation. Additionally, enclosed is a copy of the inspection report (Enclosure 5) that described the independent NRC inspection that included review of the effluent program during the 1996 operating period.

   In restarting the D.C. Cook Nuclear Plant, the NRC will conduct inspections and utilize this type of historical data and previous independent assessments to assure that effluents are controlled according to the NRC regulations and the license commitments.

2. **Regarding the visual stream of release and permissible levels of emissions:**

   a. What are the nuclides released to the air and water (i.e. the noble gases, krypton and xenon)?

   On pages A1.1-5 to A1.1-11 of the 1996 Annual Radioactive Effluent Release report (Enclosure 2) is a list of the isotopes released to air and water during 1996. You will note that not all isotopes are listed as released in every quarter. This is not uncommon as these isotopes will vary based on the reactor's power level and use of the radioactive waste processing systems.

   b. What are the half-lives/full hazardous lives?

   The half-lives of the isotopes are listed on Enclosure 4. This is a copy of the effluent isotopes documented on page A1.1-5, A1.1-8, and A1.1-9 of the 1996 effluent report. Physical and biological half-lives of these and other isotopes are publically available in a variety of reference works.

c. What are the biomedical effects?

At the dose levels that have been reported by the plant, which are within NRC's limits, no biological effects are expected.

The NRC has established limits for the release of radioactivity during routine operations. These limits are based on guidelines set by the Environmental Protection Agency and by national and international standard setting groups. Although the effects of very low levels of radiation are difficult to detect, the NRC limits are based on conservative assumptions that radiation exposures to members of the public should only be a small fraction of what they receive from natural background radiation.

Experience has shown that nuclear plants typically release only a small portion of the NRC limits during normal operation. A person spending a full year at a nuclear plant boundary would receive a radiation exposure of less than 1 percent of the radiation exposure everyone receives from natural background radiation (Natural background radiation averages 300 millirems per year, while the exposure at the plant boundary is typically around 1 to 2 millirems or less).

The licensee concluded, based on measurements at eleven offsite background stations, that dose due to direct radiation is negligible and the maximum cumulative dose to an individual from liquid and gaseous effluents during 1996 was well within the NRC limits. NRC Report Nos. 50-315/97014(DRS); 50-316/97014(DRS) (Enclosure 5) concluded that the Radiological Environmental Monitoring Program (REMP) was effective and there was no discernable impact on the environment from plant operations. The licensee followed the Off-site Dose Calculation Manual (ODCM) (Enclosure 3) methodology in collecting, processing and analyzing environmental samples and audits of the ODCM program have been effective in identifying and correcting minor procedural discrepancies as was identified in the report.

d. What are the environmental effects?

Based on the 1996 Annual Radioactive Effluent Release report (Enclosure 2), there was no discernable impact on the environment from plant operations during its last full year of operation. This report contains data on the licensee's environmental monitoring program which includes a diverse collection of environmental media that have the potential to contain licensed radioactive material released from the plant. Sampling of the air, water, sediment, milk, food products, and fish and invertebrates is performed. The water sampling includes samples of lake surface water and drinking water. All of these samples are analyzed for the presence of radioactive material. At the low levels that are being looked at for this program; please note that it is typical to observe radioactive fallout material from prior atmospheric weapons testing.The radiological environmental monitoring program required by the NRC is sufficiently comprehensive to provide an adequate assessment of the potential radiological impact of plant operation on the off-site environment. You may refer to pages A1.2-1, A1.3-1, A1.4-1 and A1.5-1 of the enclosed 1996 Annual Radioactive Effluent Release report for supporting data.

e. What are the agricultural effects on soil, grasses, and crops?

See the above answer to Question 2d.

f. When are they (the isotopes) monitored?

The ODCM describes the D.C. Cook plant monitoring program for air and water that leaves the plant. Batch releases of both air and water are monitored prior to their release. The D.C. Cook effluent monitoring program is described in detail in Sections 4.2 and 4.3 of Enclosure 3, Pages 8-17. Additionally, the Radiological Environmental Monitoring Program (REMP) is described in Section 4.5 of Enclosure 3, pages 18-19.

g. How are they (the isotopes) monitored?

The mechanics of the Radiological Effluent Monitoring Program (REMP) are described in the enclosed ODCM, Sections 4.2 to 4.5 of Enclosure 3. Environmental monitoring stations, locations and operability assessment is described in the Enclosure 3.19 of the ODCM. The NRC reviews the operability of this monitoring equipment during our independent inspections of the REMP program. An example of NRC REMP report findings is available in Enclosure 5.

h. How often are they (the isotopes) monitored?

The isotopes are monitored continuously for some pathways and periodically for others as described in the ODCM, Attachment 3.19. The Radiological Effluent Monitoring Program is described in Section 4.5 of the ODCM. These monitoring results are used to prepare the Annual Radioactive Effluent Release report (Enclosure 2).

i. Where are they (the isotopes) monitored?

The locations of monitoring, both batch and continuous release pathways, are described in the ODCM Sections 4.2 to 4.5 and specifically in ODCM Attachment 3.19.

j. By whom are they (the isotopes) monitored?

The licensee conducts the monitoring activities both on-site and at the environmental monitoring points outlined in the ODCM. The NRC conducts independent inspections of the monitoring program to assure that the equipment is properly calibrated and operable and that the samples are taken properly. Additionally, the NRC assures that the licensee's quality control program is in place to assure accurate measurement between inspections.

k. Who is informed of the releases?

The NRC is notified annually by way of the Annual Radioactive Effluent Release report. The ODCM also outlines additional reporting requirements in Section 4.8. Further, 10 CFR Part 20 describes reporting requirements to be followed if certain limits are exceeded. When these documents arrive at the NRC they become public documents and are available to the public, and most recently, via the World Wide Web. Therefore, releases are reported to the public.

l. What monitoring and results has Cook registered (informed the NRC and public)?

The D.C. Cook plant has provided the Annual Radioactive Effluent Release report to the NRC and the public since prior to the first start-up of the plant through December 1999, as described above.

m. What are the restart related plans for monitoring the isotopes?

The licensee is required to continue to monitor the environment and all discharges as it has in the past and as described in the enclosed ODCM. The NRC will continue to conduct specific independent inspection activities related to air and water effluent monitoring from D.C. Cook Nuclear Power Plant to assure adequate protection of the public health and safety and the environment. These activities will continue even after restart of the plant.

n. What projection on the above (isotopes released) are made for the Cook restart?

The NRC technical staff expects that the monitoring results shown in the 1996 Annual Radioactive Effluent Release report would be very similar to the monitoring results when the D.C. Cook plant is restarted. As was stated in NRC Report Nos. 50-315/97014(DRS); 50-316/97014(DRS) (Enclosure 5), Tritium effluents are expected to decline significantly due to new steam generators being installed.

o. What are the plans for public awareness and protection?

The NRC is a public agency and all non-proprietary information, like the environmental monitoring data and all estimates of dose to members of the general public, are publically available. The licensee has made public its schedule and projected start-up dates through periodic press releases. The NRC will publicly announce the closure of the D.C. Cook Restart Action Plan prior to restart of Unit 2. As noted above, the NRC will continue to independently review the D.C. Cook environmental monitoring program.

p. What is the Cook radioactive waste storage situation?

NRC Inspection Report Nos. 50-315/99005(DRS); 50-316/99005(DRS) (Enclosure 6) includes a review of the Dry Active Waste (DAW) management program. The conclusion was that overall, the solid radioactive waste management program was effectively implemented. Section R2 of the report states, "Over the last several years, the licensee significantly reduced the back log of stored waste that accrued while the State of Michigan licensees were banned from the Barnwell disposal site. Only a few HICs (High Integrity Containers) containing dewatered resins remained in the Radioactive Material Storage Building (RMB) and were awaiting shipment to the Barnwell site." While the report does not specifically address the total volume of space available for onsite storage of DAW, the licensee did successfully house DAW for several years with no apparent detriment. The inspectors did find the material condition of the facilities excellent.

Regarding spent fuel storage, the licensee currently utilizes a spent fuel pool. The licensee has not submitted plans for use of dry cask storage of spent fuel at this time. They have not indicated to the NRC that they are pursuing this option at this time.

q. What is the radioactive waste storage capacity for the future?

See the above answer to Question 2p.

r. What are the cask storage off-load plans (and data)?

See the above answer to Question 2p.

3. **Is/was the NRC authorized to fine up to $110,000 for each of the 44 uncorrected violations since Three Mile Island requirements to take plant initiatives?**

The NRC was authorized the in Energy Reorganization Act of 1974 to promulgate rules and regulations, this includes the authorization to assess civil penalties. The rules for use of civil penalties are contained in 10 CFR 2.205. The NRC Office of Enforcement Administers the Enforcement Policy as outlined in NUREG-1600. A copy of the most recent version of NUREG-1600 is enclosed for your review (Enclosure 7).

A review of the NRC's enforcement actions involving D.C. Cook reveals action taken on October 13, 1998, as a result of inspection findings from five inspections conducted from August 4, 1997 to April 15, 1998. As a result, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalty. A copy is enclosed as Enclosure 8. Using the NRC Enforcement Policy, the NRC assessed a $500,000 fine for the 38 violations cited and other apparent violations described in the inspection reports. In addition to this action, the licensee was required to provide evidence of the corrective actions. During the two year outage at D.C. Cook, the licensee implemented corrective action for these findings in addition to repairing and replacing system components for continued safe plant operation**.**

How can the public accept such past irresponsibilities and current/future plans?

The NRC, through the resident inspector program, has inspection staff on site consistently through out the work week (including some weekends) and at varied times of the day to assure that problems in plant equipment, personnel performance and operator activities can be readily identified and corrective actions promptly implemented. The licensee, during this outage, has significantly improved their own self-assessment and corrective action program to more readily identify problems and effectively implement corrective actions. The NRC has been in constant review of this process improvement and will continue to independently monitor the licensee performance.

4. **What monitoring data is shared with local authorities (i.e., State Police, Emergency Management Organizations)?**

The Michigan State Department of Public Health directly receives copies of the Annual Radioactive Effluent Release reports. The emergency response organizations have the information available to them upon request and as publically available documents. The emergency response organizations typically are interested in effluents related to emergency conditions and have less need for normal operational effluent data.

Enclosure 5

September 1, 1997

Mr. E. E. Fitzpatrick
Executive Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NRC ROUTINE CHEMISTRY AND RADIATION PROTECTION INSPECTION REPORT NOS. 50-315/97014(DRS); 50-316/97014(DRS)

Dear Mr. Fitzpatrick:

This refers to the inspection conducted by D. Hart and R. Paul of this office on August 4-8, 1997. The inspection included a review of authorized activities for your D. C. Cook, Units 1 and 2 Reactor Facilities.

The inspection was an examination of activities conducted under your license as they relate to chemistry, to radiation safety, and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

Overall, the chemistry and radiological environmental monitoring programs were well implemented. The inspectors observed that chemistry technicians were knowledgeable in sampling and analysis procedures, and noted that corrective actions were being implemented in response to an audit of the chemistry program. Interlaboratory program results were generally good; however, a weakness was identified by the inspectors in that there was no tracking or trending of comparison results to determine overall performance, no documentation of the acceptance criteria used, and no instructions were given as to when sample re-analysis is required.

An audit of the radiological environmental monitoring program identified issues which were corrected by your radiation protection department. Also the inspectors were able to observe training provided to your radiation protection technicians regarding recent health physics industry events. This training was a good initiative on the part of the training and health physics staff.

In accordance with 10 CFR 2.790 of the NRC'S "Rules of Practice," a copy of this letter and the enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

/s/ J. A. Grobe

John A. Grobe, Acting Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-315/97014(DRS); 50-316/97014(DRS)

cc w/encl: A. A. Blind, Site Vice President
John Sampson, Plant Manager
James R. Padgett, Michigan Public
Service Commission
Michigan Department of Environmental Quality

Mr. E. E. Fitzpatrick
Executive Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
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John A. Grobe, Acting Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-315/97014(DRS); 50-316/97014(DRS)

cc w/encl: A. A. Blind, Site Vice President
John Sampson, Plant Manager
James R. Padgett, Michigan Public
Service Commission
Michigan Department of Environmental Quality

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos: License Nos:	50-315; 50-316 DPR-58; DPR-74
Report Nos:	50-315/97014(DRS); 50-316/97014(DRS)
Licensee:	American Electric Power Company
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	500 Circle Drive Buchanan, MI 49107-1395
Dates:	August 4-8, 1997
Inspectors:	R. Paul, Senior Radiation Specialist D. Hart, Radiation Specialist
Approved by:	Gary L. Shear, Chief, Plant Support Branch 2 Division of Reactor Safety

EXECUTIVE SUMMARY

D. C. Cook Units 1 and 2
NRC Inspection Report 50-315/97014; 50-316/97014

This inspection included a review of the chemistry program and the Radiological Environmental Monitoring Program (REMP). The following specific observations were made:

Plant Support

- The primary and secondary systems water chemistry was well maintained and monitored. The licensee took appropriate actions to investigate and correct any adverse trends identified (Section R1.1).

- The laboratory and in-line instrument quality control program was well implemented and ensured the accuracy of chemistry instrumentation. Interlaboratory program results were generally good; however, a weakness was identified by the inspectors in that there was no tracking or trending of comparison results to determine overall performance, no documentation of the acceptance criteria used, and no instructions for when to re-analyze samples (Section R1.2).

- The inspectors reviewed the post accident sampling system (PASS) program to ensure operability. In addition, the inspectors reviewed maintenance records and discussed system operability with cognizant chemistry staff. No problems with the PASS were identified (Section R1.3).

- Implementation of the REMP was effective and no discernable impact on the environment from plant operations was identified. Tritium from the absorption pond continued to be tracked with no evidence that tritium had reached drinking water in either St. Joseph or Lake Township (Section R1.4).

- The inspectors observed a chemistry technician (CT) sample primary coolant. The CT demonstrated good contamination control practices as well as good analytical techniques and knowledge of procedure requirements (Section R4.1).

- The inspectors observed training provided to the radiation protection technicians (RPT's) regarding recent industry events. Good interaction between the instructor and the RPT's was observed and questions asked were appropriate (Section R5.1).
- Audits of the chemistry and radiological environmental monitoring programs identified several areas with minor deficiencies. Corrective actions had been implemented for the identified deficiencies (Section R7.1).

Report Details

IV. Plant Support

R1 Radiological Protection & Chemistry (RP & C) Controls

R1.1 Plant Water Chemistry Control

a. Inspection Scope (84750)

The inspectors reviewed the management of primary and secondary water chemistry including the program to mitigate impurities in the systems. The trending and evaluation of chemistry parameters by chemistry supervision was also reviewed.

b. Observations and Findings

The water chemistry program was consistent with the Electric Power Research Institute (EPRI) pressurized water reactor guidelines. A review of selected trend records indicated that plant primary water quality was very good, and no significant problems were observed. Once trends with the chemistry parameters were identified the chemistry department took prompt action to identify and correct the cause. None of the station chemistry department's goals had been exceeded for primary water chemistry for 1997.

The secondary water chemistry had periodically exceeded the station limits for sodium (Na), and condensate oxygen. The increased levels of Na seen on both units were attributed to secondary equipment being placed back into service after maintenance. The chemistry department had tracked these instances and issued a condition report for an adverse trend. Chemistry staff recommended that maintenance evaluate the use of steam cleaned parts prior to installation into the system, evaluate the effectiveness of cleaning after maintenance, flushing of the systems involved, and that chemistry perform sampling prior to placing equipment back into service. Chemistry staff also recommended revision of several operations procedures to include filling and draining systems prior to returning them to service.
The increase in condensate oxygen had exceeded the EPRI action level 1 limits for both units. This increase was attributed to the station performing condenser leak checks and high lake temperatures have also caused the levels to increase on Unit 1. The station plans to put a polymer coating on the condenser tube sheets from the water box up to the tube sheet gasket area during the next Unit 2 outage to help reduce inleakage thus decreasing oxygen concentrations.

Ethanolamine (ETA) is utilized for pH control and to reduce iron transport. Chemistry had noted high "after" cation conductivity values caused by the ETA, and through discussions with other stations chemistry personnel decided to test the viability of shutting off ETA addition after the initial treatment. At operating temperature, pH and iron transport were being closely monitored to ensure they remained within the appropriate parameters.

NRC Report No. 96004 noted that the reverse osmosis makeup water purification system did not have an output capacity to meet the demands of both units. Since that observation a new permanent system had been installed with two trains, each capable of processing 800 gallons per minute, which is sufficient to meet station needs.

c. Conclusions

The primary and secondary water quality was well maintained and monitored. The chemistry staff took appropriate actions to investigate and correct any adverse trends.

R1.2 Quality Control of Laboratory and In-line Chemistry Instruments

a. Inspection Scope (84750)

The inspectors reviewed the licensee's quality control (QC) program for laboratory and in-line instruments, radiochemistry instrumentation, and the interlaboratory comparison program. The inspectors reviewed the licensee's implementation of procedure 12 THP 6020 ADM.001, "Quality Control," Revision 1, dated June 17, 1996, and the maintenance of instrument control charts and performance of instrument calibrations.

b. Observations and Findings

The inspectors reviewed the labeling and storage of reagents and calibration standards. The inspectors did not identify any chemicals which were improperly labeled or which had been used beyond their expiration date. Laboratory chemicals were appropriately stored (i.e., incompatible chemicals were not stored in common locations).

The inspectors observed that performance tests for the laboratory, radiochemistry, and in-line instruments were accomplished. The laboratory control charts were well maintained and indicated proper instrument response, and statistical distribution of performance test data. The in-line instruments were tested as required with corrective actions taken for instruments not meeting the stated acceptance criteria. The inspectors also reviewed the efficiency determination for different geometries and compared them to the calibration curves for the high purity germanium detectors. The generation of lower limits of detection for the germanium detectors was also reviewed, and no problems were identified.

The inspectors noted that the station participated in the NWT Corporation interlaboratory comparison program and had generally performed well. However, the inspector identified that there was no tracking or trending of the results to determine if any analyte may have been consistently missed, or if any other problems may exist. Also, the acceptance criteria for the testing was not documented in any station procedure. NWT Corporation identified the acceptance criteria used by the Institute for Nuclear Power Operations (INPO) as the standard they used to determine if a plant was within tolerance or not. When the chemistry department had failed to meet the INPO acceptance criteria the chemistry staff attempted to identify reasons and performed re-analysis using the duplicate samples which had been provided by NWT. These results were reported in the chemistry monthly report; however there was nothing in station procedures to direct personnel to re-analyze or report the results. Although this had been done as a good laboratory practice, the lack of any procedural guidance contributed to the weakness in the interlaboratory program.

c. <u>Conclusions</u>

The laboratory and in-line instrument quality control program was well implemented and ensured the accuracy of chemistry instrumentation. Interlaboratory program results were generally good; however, a weakness was identified by the inspectors in that there was no tracking or trending of results to determine overall performance, no documentation of the acceptance criteria used, and no procedural guidance for re-analyzing a sample.

R1.3 <u>Post Accident Sampling System Maintenance and Surveillance Program</u>

The inspectors reviewed the post accident sampling system (PASS) to ensure operability, reviewed maintenance records, and discussed system operability with cognizant chemistry staff.

The inspectors noted that chemistry personnel had a thorough understanding of the PASS system including the process with which a sample is obtained, system connections, and the maintenance history. The material condition of the PASS stations was good and the licensee was capable of obtaining required samples.

The inspectors noted that the licensee had effectively maintained the material condition of the PASS system to ensure the capability of sampling during accident conditions.

R1.4 <u>Radiological Environmental Monitoring Program</u>

a. <u>Inspection Scope (84750)</u>

The inspectors reviewed the implementation of the Radiological Environmental Monitoring Program (REMP) based on requirements of the licensees Off-site Dose Calculation Manual (ODCM). The inspectors also observed air and drinking water collection and examined air sampling equipment. The 1995 and 1996 Annual Environmental Operating Reports (AEORs) were reviewed to ensure that the reports were submitted as required and to evaluate the effect of the plant’s operations on the environment.

b. <u>Observations and Findings</u>

The REMP data indicated that plant operation had no radiological impact on the environment. In addition, the REMP staff conducted sampling and analyses according to technical specifications and all deviations were appropriately noted. The material condition of the air sampling equipment was very good and was within calibration. All air samplers were operational and sampling activities were performed in accordance with station procedures.

Plant personnel continued to track and trend the movement of groundwater tritium. The tritium in the groundwater at the plant has been periodically reviewed by NRC inspectors. The source of the tritium is from primary to secondary leakage primarily in Unit 1, which flows from the turbine room sump into the absorption pond. Seepage from the pond was evaluated by the licensee in a 1991 Hydrogeologic Evaluation, the results of which determined that natural barriers in the environment would prevent the tritium from reaching a source of drinking water. The groundwater well located adjacent to the absorption pond (w-14) recently showed tritium levels up to 19,000 pCi/l. The plant has a reporting level of 20,000 pCi/l, and plant staff stated that any values in excess of the reporting level would be mentioned in the AEOR. No tritium above background levels had been found in drinking water analyzed for St. Joseph and Lake Township. Unit 1 is scheduled for a steam generator replacement in 2000, which should greatly reduce the primary to secondary leakage, and subsequently the tritium in the absorption pond.

c. Conclusions

Implementation of the REMP was effective and no discernable impact on the environment from plant operations was identified. Tritium from the absorption pond continued to be tracked with no evidence that tritium had reached drinking water in either St. Joseph or Lake Township.

R4 Staff Knowledge and Performance in RP&C

R4.1 Sampling and Analysis of Primary Coolant

The inspectors observed a chemistry technician (CT) sample primary coolant. The technician appropriately contacted the control room prior to obtaining the sample and at the end of sampling. The CT demonstrated good contamination control practices, as well as good analytical techniques and knowledge of procedure requirements. The inspectors noted that during analysis, when the CT could not find a graph that he was directed to go to in the procedure, he stopped and notified management as appropriate. The graph was found, it had not been attached to the current revision; to correct this oversight, chemistry supervision immediately revised the procedure to include the graph.

R5 Staff Training and Qualification in RP&C

R5.1 Industry Events Training for Radiation Protection Personnel

The inspectors attended training provided to the radiation protection technicians (RPT's) regarding recent industry events. Events reviewed included the Calvert Cliffs incident involving a diver in the spent fuel pool and the intake of alpha emitting nuclides by a worker at Haddem Neck. Good interaction between the instructor and the RPT's was observed. The training was performed in a workshop format, groups were formed to evaluate these events, and present their findings to the class.

R7 Quality Assurance in RP&C Activities

R7.1 Chemistry and REMP Audits

The inspectors reviewed the results of a quality assurance audit of chemistry, QA-96-16. Several condition reports were issued as a result of the auditor's findings. The inspectors noted that the audit was comprehensive and that corrective actions and recommendations were being implemented.

The inspectors also reviewed the last two REMP audits, QA-97-18 and QA-96-10. The most recent audit had no findings in the REMP area, however there were several good findings in the previous audit. A discrepancy between the ODCM and a procedure was identified during this audit and a condition report was issued. The procedure was revised to reflect the ODCM requirements.

R8 Miscellaneous RP&C Issues

R8.1 (Closed) Violation No. 50-315/96004-02; 50-316/96004-02: This violation had three examples, the first was a failure to perform monthly grab samples for the PASS comparisons. The chemistry staff was directed to count the samples the same day they were taken in order to address this issue. This was communicated to the staff through department meetings, no recurrence of the previous problems were identified by the inspectors. The second example was a failure to take corrective actions for the comparisons outside of the acceptance criteria. Procedure 12 THP 6020 PAS.016 was revised to include steps directing the worker to re-analyze a PASS sample if outside the acceptance criteria, and if the re-analysis was outside the limits, discussions with PASS supervision were conducted to record deviation in appropriate logbook or database. The lack of procedural adherence concerning the evaluation and documentation of quality control information was the third example and was addressed by revising the stations procedures to clarify steps for identifying, recording, and evaluating biases. The inspectors reviewed the corrective actions and concluded that the actions were appropriate.

V. Management Meetings

X1 Exit Meeting Summary

On August 8, 1997, the inspectors presented the inspection results to licensee management. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

M. Ackerman, Nuclear Licensing
T. Andert, Chemist
R. Claes, Chemist
D. Foster, Health Physicist
R. Gillespe, Acting Plant Manager
M. Snyder, Health Physicist

INSPECTION PROCEDURES USED

IP 83750: Occupational Radiation Exposure
IP 84750: Radioactive Waste Treatment, and Effluent and Environmental Monitoring

LIST OF ITEMS OPENED AND CLOSED

Closed

50-315(316)/96004-02 VIO failure to follow procedures with regard to PASS monthly comparisons, corrective actions for the comparisons, and failure to document and evaluate QC information.

LIST OF ACRONYMS USED

AEOR	Annual Environmental Operating Report
CT	Chemistry Technician
EPRI	Electric Power Research Institute
ETA	Ethanolamine
INPO	Institute for Nuclear Power Research
ODCM	Off-site Dose Calculation Manual
NRC	Nuclear Regulatory Commission
PASS	Post-Accident Sampling System
PDR	Public Document Room
QC	Quality Control
REMP	Radiological Environmental Monitoring Program
RPT	Radiation Protection Technician
VIO	Violation

LIST OF DOCUMENTS REVIEWED

Annual Environmental Operating Report 1996

Annual Environmental Operating Report 1995

12 THP 6020 ADM.001 Rev 1 “Quality Control”

12 THP 6020 ADM.010 Rev 2 “Analytical Results”

12 PMP 6010 OSD.001 Revision (Rev) 11 “Offsite Dose Calculation Manual”

12 THP 6010 RPP.632 Rev 4 “Collection of Environmental Air Samples”

12 THP 6010 RPP.642 Rev 1 “Collection of Drinking Water Samples”

12 THP 6020 CHM.202 Rev 3 “Condensate and Feedwater”

12 THP 6020 PAS.016 Rev 4 “Post Accident Sampling Quality Control”

Performance Assurance Audit QA-97-18 “Radiological Environmental Monitoring Program (REMP) / Offsite Dose Calculation Manual (ODCM) (PMI-6010)”

Performance Assurance Audit QA-96-10 “REMP (PMI-6010)

Performance Assurance Audit QA-96-16 “Chemical/Radiochemical Control Program (PMI-6020)”

NWT Result Nos. 47, 48, 49, and 50

Condition Report Nos. 96-0838, 97-1644, 97-1796, 97-0063, 97-0256, 97-0654, and 97-0655

Technical Specification Sections 6.8 and 3.4.7

Updated Final Safety Analysis Report Sections 2.7 and 9.6

Enclosure 6

April 9, 1999

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

SUBJECT: NRC ROUTINE RADIATION PROTECTION INSPECTION REPORTS 50-315/99005(DRS); 50-316/99005(DRS)

Dear Mr. Powers:

On March 19, 1999, the NRC completed a routine inspection at your D.C. Cook Units 1 and 2 reactor facilities. The enclosed report presents the results of this inspection.

The inspection was an examination of activities conducted under your license as they relate to radiation safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Specifically, the inspection focused on the solid radioactive waste (radwaste) management and transportation programs, and included a review of the quality assurance program for the use of NRC approved packages and the audit and self-assessment program relative to the areas reviewed. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

No violations of NRC requirements were identified. Overall, the solid radwaste management and transportation programs were effectively implemented. Radioactive waste streams were processed into solid forms in accordance with the Process Control Program (PCP) and adequately trained staff ensured that wastes were characterized, packaged, and shipped in accordance with NRC and Department of Transportation (DOT) requirements and station procedures. Supervisory oversight of station radwaste staff and vendor processing activities was effective, and audits were generally properly focused and were of sufficient scope and depth to assess program performance. However, some deficiencies were identified with the development of the procedure for the waste classification scaling factor program. Also, deficiencies were identified with the level of detail in the PCP concerning 10 CFR 61 waste characteristic requirements and with the consistency between the PCP, the Final Safety Analysis Report and the current waste processing program. Your staff acknowledged these deficiencies and was considering actions to address them.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public document Room.

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

Sincerely,

Original /s/ J. A. Grobe

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Reports 50-315/99005(DRS); 50-316/99005(DRS)

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

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

Sincerely,

John A. Grobe, Director
Division of Reactor Safety

Docket Nos.: 50-315; 50-316
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report Nos:	50- 315/99005(DRS); 50-316/99005(DRS)
Licensee:	American Electric Power Company
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	March 15 -19, 1999
Inspectors:	W. Slawinski, Senior Radiation Specialist A. Kock, Radiation Specialist
Approved by:	Gary L. Shear, Chief, Plant Support Branch Division of Reactor Safety

EXECUTIVE SUMMARY

D.C. Cook, Units 1 and 2
NRC Inspection Reports 50-315/99005; 50-316/99005

This routine, announced inspection assessed the effectiveness of the licensee's solid radioactive waste (radwaste) management and radioactive material transportation programs, and included a review of the quality assurance program for NRC approved packages and the audit and self-assessment program. Within these areas, the following conclusions were made:

Plant Support

- Direct licensee oversight of vendor resin dewatering activities and extensive supervisory involvement in radwaste processing ensured effective implementation of the radwaste management program. Wet solid wastes were processed in accordance with the Process Control Program (PCP) and implementing procedures, and dewatered waste streams were properly sampled and verified to ensure that regulatory limits for free standing liquid were met (Section R1.1).

Deficiencies were identified with the level of detail in the licensee's PCP concerning 10 CFR 61.56 waste characteristics, and both the PCP and the Final Safety Analysis Report (FSAR) were not fully consistent with current onsite waste processing activities (Section R1.1).

- The radwaste packaging and transportation program was effectively implemented. Shipments of radwaste were appropriately classified, vehicle and package surveys were performed as required, and manifests were completed in accordance with regulatory requirements (Section R1.2).

The program for classifying waste streams and scaling difficult to measure radio-nuclides was implemented in accordance with station procedures and industry guidance. However, the procedure for the scaling factor program was not sufficiently developed to ensure consistent and appropriate implementation of the program (Section R1.2).

- The training provided to licensee staff involved in the preparation and shipment of radioactive materials satisfied Department of Transportation requirements and imparted an adequate level of knowledge to ensure effective program implementation (Section R5.1).

- The audit and self-assessment programs for the packaging and transportation of radioactive material for the 10 CFR 71 Quality Assurance program and for the processing of radwaste were effectively implemented. Audits and assessments were generally properly focused and were of sufficient scope and depth to assess program performance. Identified deficiencies were placed into the licensee's corrective action process for resolution (Section R7.1).

DETAILS

IV. Plant Support

R.1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radioactive Waste (Radwaste) Processing

a. Inspection Scope (IP 86750)

The inspectors reviewed the licensee's solid radwaste management program including the Process Control Program (PCP) and associated implementing procedures for the processing of radwaste, and the licensee's oversight of processing activities.

b. Observations and Findings

The licensee's solid radwaste streams consisted of spent resins from primary systems, secondary resins from radwaste processing systems, filter cartridges and filter media and various types of contaminated dry wastes (Dry Active Waste (DAW)).

Spent resins from primary systems were transferred to the spent resin storage tank by the licensee and subsequently sluiced into High Integrity Containers (HICs) and dewatered onsite by a vendor. Dewatered resins were subsequently shipped to a licensed waste disposal site or transferred to a vendor for further processing prior to disposal at the burial site. Filter cartridges and media were typically stored onsite and air dried and/or dewatered, and subsequently shipped for disposal to a licensed burial site. Dry active wastes and potentially contaminated solid wastes generated in radiologically protected areas were placed in designated receptacles throughout the plant, collected and sorted by the licensee, and transferred to a vendor for compaction and consolidation prior to disposal at a burial site. 10 CFR 61 waste stability requirements for dewatered waste products were met by placing the processed waste in a disposal container (e.g., HIC) that provided stability for land burial. A waste evaporator system previously used to concentrate dissolved or suspended solids in liquid radwaste streams has not been used for several years and was no longer maintained. Similarly, waste solidification processes ceased several years ago, although vendor solidification remained as an option for achieving waste stability.

The licensee developed a PCP to establish the parameters and provide assurance that the processing of radwaste resulted in a waste form that meets the requirements of both 10 CFR 61 and the low level waste disposal site licenses. The inspector reviewed the latest revisions to the licensee's PCP and the Final Safety Analysis Report (FSAR), and identified deficiencies with the PCP and inconsistency between both the PCP and FSAR compared to current radwaste processing activities. Specifically, both the PCP and FSAR referred to use of a waste evaporator system to process liquid wastes, and specify that solidification processes were used to achieve waste stability for certain waste forms. Additionally, the PCP described the solidification process and specified that written procedures are maintained to address solidification testing. However, because solidification processes ceased several years ago, solidification procedures were no longer maintained. Also, while the PCP adequately addressed the free standing liquid requirements of 10 CFR 61.56, other waste characteristic requirements such as those for chemical and biological hazards were not addressed in the PCP. Although the licensee implemented programs to limit and control the use of chemicals at the station and the licensee periodically monitored liquid radwaste streams to evaluate chemical and environmental hazards, the licensee agreed that the PCP should address all the waste characteristic requirements of 10 CFR 61.56 to ensure consistency with the NRC's Branch Technical Position on Waste Form. The licensee planned to address the deficiencies with the PCP and evaluate the inconsistencies between the FSAR and PCP and revise these documents as warranted.

Vendor waste processing activities were governed by station procedures and implemented under the supervision of the licensee's radwaste staff. The inspectors reviewed the licensee's procedures for resin transfer, resin dewatering and for testing of the processed product, and concluded that the procedures were clear and included acceptance criteria that were consistent with 10 CFR 61.56 requirements for free standing liquid in the final waste form. The licensee's radwaste handling group provided oversight of contractor resin sluicing and dewatering activities and effectively monitored vendor activities by direct observation, supervisory review of waste processing records, and independent verification that dewatered products met free standing liquid requirements. The radwaste group packaged and/or completed final preparations for all shipments. Group supervisors ensured that packaging and shipment activities were completed in accordance with regulatory requirements and station procedures. Inspector discussions with those licensee staff involved in waste processing and inspector review of dewatering data revealed that the staff was knowledgeable of vendor activities and that waste was processed in accordance with station procedures. Dewatering calculations for several waste shipments were reviewed by the inspectors and confirmed that the waste met the 10 CFR 61 free standing liquid requirements. The inspectors concluded that the licensee exercised good oversight of contractor onsite dewatering activities and that radwaste group supervisors were extensively involved in each radwaste shipment.

c. <u>Conclusions</u>

Direct licensee oversight of vendor onsite resin dewatering activities and extensive supervisory involvement in radwaste processing ensured effective program implementation. Wet solid wastes were processed in accordance with the PCP and implementing procedures. Dewatered waste streams were sampled and verified to ensure that regulatory limits for free standing liquid were met. The licensee's PCP, however, was not sufficiently detailed to address all pertinent 10 CFR 61.56 waste characteristic requirements, and both the PCP and FSAR were not fully consistent with waste processing activities.

R1.2 Radwaste Packaging and Transportation

a. Inspection Scope (IP 86750)

The inspectors reviewed the licensee's radwaste packaging and transportation program for compliance with NRC, Department of Transportation (DOT) and waste burial site requirements, and evaluated the licensee's waste stream scaling factor (classification) program. The review included interviews of plant staff and review of station procedures, scaling data, and records of past shipments.

b. Observations and Findings

The inspectors verified that the licensee maintained current copies of NRC and DOT regulations, burial site and waste processor licenses, and that staff involved in radioactive material shipments were knowledgeable of their content. The radwaste handling group coordinated shipping activities and was responsible for the radioactive material and radwaste transportation program. A radioactive material senior specialist and the radwaste handling supervisor provided oversight of the radioactive material transportation program and ensured, by direct involvement, that the program was implemented in accordance with regulatory requirements and station procedures.

The inspectors reviewed station procedures governing radioactive material/radwaste packaging and shipping, completion of shipment manifests and for radwaste classification, and discussed their implementation with plant staff. The procedures reviewed by the inspectors were technically accurate and generally consistent with DOT and NRC requirements; however, some minor problems were identified with the procedure for the preparation of radioactive shipments, involving package marking and shipment manifest information. These deficiencies did not result in mislabeled or improperly completed manifests. The procedural issues were discussed with radiation protection (RP) management, who indicated that the procedure would be revised to correct the deficiencies.

The licensee used a vendor's computer program (i.e., D. W. James and Associates) to classify waste pursuant to 10 CFR 61.55, to determine reportable quantity (RQ) values, and to generate shipping manifests. The inspectors verified that the computer program methodology for determining waste classification was consistent with 10 CFR 61 requirements.

The licensee implemented a scaling factor program for waste stream sampling and analysis for Difficult To Measure (DTM) radionuclides in accordance with the NRC Branch Technical Positions on Waste Classification and Waste Form. Information generated in the analyses was used to classify waste, as required by 10 CFR 61. Representative waste streams for DAW, filter cartridges, primary resins and radwaste stream resins were sampled by the licensee throughout a given year, and scaling factors were determined annually and compared to the previous year's data. Analyses of DTM radionuclides were contracted to a vendor laboratory, and scaling factors were prepared by the licensee using the D. W James and Associates 10 CFR 61 Sample Analysis Program. Individual radioisotopes present in a waste stream were scaled to cobalt-60 gamma emitters, except for iodine-129 and transuranic isotopes, which were scaled to cesium-137 and plutonium-239, respectively. Electric Power Research Institute recommended generic scaling factors were used for iodine-129 and technetium-99, because the licensee lacked sufficient plant specific waste stream data for these isotopes. Also, a single set of transuranic scaling factors was applied to all waste streams because the licensee's data showed uniformity across the various waste streams generated at the station. The inspectors confirmed through record review and discussions with plant staff that waste stream analyses were completed at required intervals, and that the scaling factor program was implemented consistent with industry guidelines and the station procedure. However, the inspectors identified deficiencies with the scaling factor procedure in that it was not sufficiently developed to ensure consistent program implementation. Specifically, the procedure did not address the following important aspects of the scaling factor program:

The circumstances for scaling DTM radionuclides to cobalt-60 versus scaling to other licensee measurable nuclides (depending on fuel integrity indicators and other operational data);

The frequency and methodology for trending and reporting of reactor water chemistry data that could affect waste stream classification;

The thresholds for reevaluating scaling factors based on operational data and scaling factor variations from year to year; and

The preparation of waste stream samples to ensure they adequately represent the processed waste stream for moisture content and other characteristics.

The RP management acknowledged the inspectors' findings and planned to expand the existing procedure or develop a new procedure to address the deficiencies.

Fifty-four shipments of radwaste generated by the licensee were made to licensed burial sites in 1998. Of these 54 shipments, 15 were made from vendor facilities after additional processing. In 1997, 25 radwaste shipments were made directly to burial sites by the licensee, and 63 shipments were made to waste processors and subsequently to the burial site. The inspectors independently verified that three selected radwaste burial site shipments made in 1998 were correctly classified, that scaling factors were properly applied, that package labeling and marking was satisfactory, and that the results of package and transport vehicle surveys satisfied DOT requirements. The inspectors also verified that shipment manifests were completed consistent with the regulations and included emergency response information, and that the shipments were tracked as required by 10 CFR 20.

The licensee periodically used NRC approved (10 CFR 71) Type "A" casks for shipment of low specific activity material to licensed burial sites, as authorized by 10 CFR 71.52. The inspectors verified that valid Certificates Of Compliance (COCs) were maintained by the licensee for those casks used in 1998 to date, and that the licensee was registered with the NRC as a user of the package under the general license provisions of 10 CFR 71.12. The inspectors reviewed COCs for two recently used casks and determined that the type, form, and quantity of the material shipped in the casks by the licensee was in compliance with the certificate. The licensee recognized that the COCs for those packages approved for use pursuant to 10 CFR 71.52 would expire on April 1, 1999, and could not be renewed. Consequently, the licensee planned additional shipments in these packages before the deadline.

In 1999, the licensee planed to ship four steam generator lower assemblies, removed from Unit 2 in 1988, to the Barnwell low-level waste site for disposal. To accommodate this project, the licensee sought exemptions from the DOT for surface contaminated object packaging requirements and for transport conveyance radioactivity limits. The latter exemption request was subsequently withdrawn by the licensee after additional steam generator sampling and analysis showed that the conveyance limit would not be exceeded. The steam generators will be shipped as unpackaged, surface contaminated objects pending DOT approval. The inspectors discussed the project with the licensee's staff, and reviewed the DOT exemption request and supporting waste characterization information. The actions completed by the licensee were consistent with NRC Generic Letter 96-07, "Interim Guidance on Transportation of Steam Generators." Waste characterization work was comprehensive and technically sound. The licensee was on track for shipment of the steam generators in June 1999, after the DOT exemption was approved and final preparations were completed.

c. Conclusions

The radwaste packaging and transportation program was effectively implemented. Radwaste shipments were appropriately classified, vehicle and package surveys were performed as required, and manifests were completed in accordance with regulatory requirements. The program for classifying waste streams and scaling difficult to measure radionuclides was implemented in accordance with industry guidance and station procedures. However, the station procedure for the scaling factor program was not sufficiently developed to ensure consistent and appropriate implementation of the program.

R2 Status of RP&C Facilities and Equipment

The inspectors evaluated radiological area postings, package labeling, and the material condition of the solid radwaste processing areas in the auxiliary building and in the satellite radioactive material storage areas including the steam generator mausoleum, where the steam generators were being prepared for shipment. The drumming rooms and the radwaste demineralizer system room in the auxiliary building were properly posted and controlled, and material condition was acceptable. Satellite storage areas outside the protected area were posted and controlled in accordance with station procedures and regulatory requirements; however, some container labeling inconsistencies were identified by the licensee during an ongoing audit and were being addressed during the inspection. An inventory program was maintained for stored radioactive material and waste and regularly updated to reflect inventory changes; however, the licensee's ongoing audit identified deficiencies in its implementation, which were also being addressed during the inspection.

Over the last several years, the licensee significantly reduced the back log of stored waste that accrued while State of Michigan licensees were banned from the Barnwell disposal site. Only a few HICs containing dewatered resins remained in the Radioactive Material storage Building (RMB) and were awaiting shipment to the Barnwell site. The inspectors walked down the RMB and determined that the material condition of the facility was excellent.

Inspection Reports 50-315/97011(DRS); 50-316/97011(DRS) reported that several of the demineralizer cubicle rooms were not entered in several years and the condition of the equipment was unknown. To address this concern, the licensee monitored the cubicles remotely and did not identify any material condition issues.

R5 Staff Training and Qualification in RP&C

R5.1 Training of Staff Involved in Transportation of Radioactive Material

a. Inspection Scope (IP 86750)

The inspectors reviewed the training provided to station staff involved in radioactive material transportation activities (i.e., hazardous material (hazmat) employees). The inspectors discussed the training program with station staff; reviewed training certificates, lesson plans, and test results; and evaluated selected hazmat employee qualification criteria.

b. Observations and Findings

The licensee designated two "shipping qualified personnel" in its radwaste handling group who were approved to authorize the shipment of radwaste and radioactive material from the site. These individuals verified that packages were properly marked and labeled, that waste destined for burial site disposal was properly characterized, and that all NRC and DOT requirements were met before certifying the shipment and authorizing its release. A staff of seven technicians in the radwaste handling group was responsible for all aspects of radioactive material and radwaste processing and packaging other than dewatering activities. The technicians conducted all packaging, loading, and radiological support work incident to the shipment program.

The inspectors reviewed the training provided to selected individuals involved in both the shipping and receipt of radioactive material packages, including training for the technicians and supervisors in the radwaste handling group, the environmental affairs staff involved in the packaging and processing of mixed waste, and for storeroom workers that receive radioactive material shipments. Lesson plans and test results were reviewed for the environmental affairs staff and storeroom personnel, qualification criteria for technicians were evaluated, and course completion certificates for the shipping qualified personnel were reviewed. The inspectors' review disclosed that the training provided to these hazmat employees satisfied 49 CFR 172.704 requirements, and that recurrent training was provided at least every three years. Interviews of shipping qualified personnel revealed that they were very knowledgeable of pertinent transportation regulations.

c. Conclusions

The training provided to staff involved in packaging, preparation, and shipment of radioactive materials and radwaste satisfied DOT regulations and imparted an adequate level of knowledge to ensure effective program implementation.

R7 Quality Assurance in RP&C Activities

R7.1 Audits and Appraisals

a. Inspection Scope (IP 86750)

The inspectors reviewed the licensee's quality assurance (QA) program required by Subpart H of 10 CFR 71 for NRC approved packages and evaluated the audit and self-assessment program relative to processing, packaging, and transportation of radioactive material.

b. Observations and Findings

The licensee extended and applied its previously approved 10 CFR 50, Appendix B, QA program to packaging and transportation activities involving NRC approved packages, as authorized by 10 CFR 71.101. The 10 CFR 71 QA program was incorporated into the licensee's corporate QA program. The inspectors reviewed the licensee's QA program and concluded that it satisfied each of the applicable criteria of 10 CFR 71.103 through 71.137, as required.

The licensee regularly audited aspects of its QA program on a rolling schedule, so that all QA program elements were evaluated at intervals not to exceed 24 months. Additionally, performance audits of the radioactive material shipping and radwaste processing programs were completed every 24 months, as required by technical specifications, which also included elements of the QA program. The inspectors reviewed the results of a performance audit of the processing and transportation programs completed in April 1997, the preliminary results of a similar audit being completed in March 1999, and a self assessment of the radioactive material shipment program conducted in May 1998. The inspectors concluded that the audit and assessment activities were of sufficient scope and depth to assess overall performance of the radioactive material transportation and radwaste processing programs, and the effectiveness of the QA program as required by 10 CFR 71.137. The licensee acknowledged, however, that the audit of the QA program could be enhanced if compliance with the requirements contained in the COC for annual cask preventive maintenance (PM) completed by the cask manufacturer was verified as part of the audit. The performance assurance staff planned to consider expanding future audits to address this issue.

No regulatory compliance issues or significant weaknesses were identified by the licensee's audits, although the adequacy of the characterization for a recent shipment of contaminated equipment sent to another utility was questioned during the most recent audit. The March 1999 audit also identified inventory deficiencies and labeling inconsistencies for containers maintained in the satellite radioactive material storage areas. The latter audit also identified that condition reports (CRs) were not always generated for issues to ensure proper followup. The inspectors verified that CRs were issued to document the recent audit findings and to track their resolution, and that radwaste staff thoroughly investigated identified issues.

c. <u>Conclusions</u>

The audit and surveillance program for the packaging and transportation of radioactive material, the 10 CFR 71 QA program, and the processing of radwaste was effectively implemented. Audits and surveillances were generally properly focused, were of sufficient scope and depth to assess program performance. Identified deficiencies were placed into the licensee's corrective action process for resolution.

V. Management Meetings

XI Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management and other station staff at the conclusion of the inspection on March 19, 1999. On March 24, 1999, one of the inspectors contacted Michael Skow, a supervisor in the licensee's Performance Assurance Department, and further discussed the inspection findings relative to audits of the Quality Assurance program. The licensee acknowledged the findings presented and did not identify any of the information reviewed as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

J. C. Benedict, Performance Assurance
D. Bronicki, Radiation Protection Supervisor
J. Carlson, Environmental Affairs
D. Cooper, Plant Manager
R. Fein, Senior Technician, Radiation Protection
D. Foster, Performance Assurance
P. Holland, General Supervisor, Radiological Support
P. Hoppe, General Supervisor, Radiological Control
D. C. Kosloff, Licensing
J. Long, Radwaste Handling Supervisor
W. MacRae, Nuclear Materials
D. Noble, Chemistry/Radiation Protection Manager
B. O'Rourke, Licensing
M. Schaefer, Senior Radioactive Material Specialist
M. E. Skow, Supervisor, Internal Performance, Performance Assurance

INSPECTION PROCEDURES USED

IP 83750	Occupational Radiation Exposure
IP 86750	Solid Radioactive Waste Management and Transportation of Radioactive Materials

ITEMS OPENED AND CLOSED

None

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
COC	Certificate of Compliance
DAW	Dry Active Waste
DOT	Department of Transportation
DTM	Difficult to Measure
FSAR	Final Safety Analysis Report
Hazmat	Hazardous Material
HIC	High Integrity Container
PA	Performance Assurance
PCP	Process Control Program
QA	Quality Assurance
Radwaste	Radioactive Waste
RMB	Radioactive Material Storage Building
RP	Radiation Protection

PARTIAL LIST OF DOCUMENTS REVIEWED

Station Procedures

12 THP 6010 RPP.908, (Rev 3), Surveillance, Inventory and Inspection of Stored Radioactive Material

12 THP 6010 RPP.903, (Rev 1), Activity Determination and Waste Classification

12 THP 6010 RPP.900, (Rev 6), Preparation of Radioactive Shipments

12 THP 6010 RPP.901, (Rev 3), Resin Transfer to Qualified Shipping Container

12 THP 6010 RPP.902, (Rev 2), Dewatering of High Integrity Containers

PMP 6010 PCP.900, (Rev 3), Radioactive Waste Process Control Program

Audits, Surveillances and Related

PA Audit PA-97-05/NSDRC #241, April 1997

Audit Plan #21, PA-99-10/NSDRC #268

Surveillance #RPS-98-008

Surveillance #99-027

Other

COCs #9159, Rev 8; #222, Rev 6; and #9176, Rev 14

Waste Manifests #RMC-98-04, 1/7/98; #RMC-98-119, 10/20/98; #RMC-98-095, 9/3/98; and #RMC-98-024, 2/18/98

QA Program For The Cook Nuclear Plant, July 1995

Qualification Standards RP-0-PR00, (Rev 0), Radioactive Material Packaging and Shipping Activities

Lesson Plan HM-C-ST05, (Rev 0), DOT Hazmat Retraining

Scaling Factor Determination Report, 1997

October 13, 1998

EA Numbers 98-150, 98-151, 98-152 and 98-186

Mr. John Sampson
Site Vice President
Nuclear Generation Group
Indiana Michigan Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $500,000 (NRC Inspection Reports 50-315(316)/97201(NRR), 50-315(316)/97017(DRP), 50-315(316)/98004(DRS), 50-315(316)/98005(DRS), and 50-315(316)/98009(DRS))

Dear Mr. Sampson:

The NRC conducted five inspections at the Indiana Michigan Power (IMP) Donald C. Cook Nuclear Power Plant from August 4, 1997 through April 15, 1998. These inspections included evaluations and assessments of the: (1) ice condenser surveillance program, (2) corrective action program, (3) facility design basis, (4) safety evaluation program, and (5) control of foreign material in the containment. Because of the seriousness of the issues resulting from these inspections, lengthy public meetings were held on December 12, 1997, December 22, 1997, and January 8, 1998. The NRC held an open predecisional enforcement conference in the Region III office on May 20, 1998, with video viewing by members of the public and NRC staff in the NRC Rockville, Maryland office.

Based on the information developed during these inspections, provided during the public meetings, and provided during the predecisional enforcement conference, the NRC has determined that numerous violations of NRC requirements occurred. The circumstances surrounding these violations are described in detail in the subject inspection reports and the violations are cited in the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice). The violations have been grouped into four areas: (1) section A, performance of surveillance test activities, (2) section B, implementation of the corrective action program, (3) section C, control of the facility design basis, and (4) section D, conduct of safety evaluations.

During the predecisional enforcement conference, IMP admitted all the apparent violations that formed the basis for the conference, described its assessment of the root causes, and presented its corrective actions to address these issues. IMP stated that a root cause for many of these apparent violations was the failure to establish and communicate adequate performance standards.

As a consequence of the violations, extensive degradation of the design of each unit's containment and emergency core cooling systems (ECCS), including the ice condensers, refueling water storage tanks (RWST), and containment sumps occurred, adversely impacting the ability of both of the remaining design barriers (fuel cladding and containment) to prevent fission product release to the environment in the event of an accident. With regard to the fuel cladding barrier, deficiencies were identified involving: (1) a large quantity of fibrous materials within containment which would likely have clogged the ECCS suction strainers in the recirculation mode, (2) a single failure ECCS vulnerability, and (3) the amount of water available in the ECCS sump. With regard to the containment barrier, the effects of the degradation to the ice condenser from blocked ice bed flow passages, missing ice segments and ice basket damage represent a serious impairment of the function of the ice condenser to condense steam and suppress peak pressure. These conditions resulted in a serious impairment of the safety function for all redundant trains of ECCS and for containment. Further, beyond the specific systems addressed by this enforcement action, two additional systems related to the containment, the hydrogen ignition and containment spray systems, were also degraded during the same period and following analysis the licensee declared these systems inoperable.

The eight violations in section A of the Notice demonstrate that the surveillance program intended to ensure the continued availability of safety systems was inadequate. Procedures implemented to ensure post refueling outage containment cleanliness inspections were inappropriate as demonstrated by the thousands of pounds of debris present in containment for several operating cycles. The debris, which consisted of insulation, coatings (paint), labels, tape, and granular charcoal would, during a loss of coolant accident (LOCA), deposit on suction strainers used for long-term recirculation cooling and significantly impede reactor core cooling. Several procedures implemented for ice condenser testing were inadequate as demonstrated by (1) visual examinations that failed to detect excessive ice blockage of ice condenser flow passages, (2) acceptance criteria that failed to account for measurement errors, and (3) the selection of a population of baskets to weigh that was not representative of conditions within the ice condenser. In addition to the procedure problems, IMP failed to monitor the quality of services provided by contractors performing ice condenser surveillance activities and to detect rough handling practices that caused structurally significant ice basket damage to go undetected. These violations represent a programmatic breakdown in the control of IMP’s surveillance program for the ice condenser.

The six violations in section B demonstrate a failure of the Donald C. Cook corrective action program to promptly identify significant conditions adverse to quality, to take appropriate corrective actions to determine the cause of each condition, and implement corrective actions to preclude repetition. For example, dented/buckled ice basket webbing and missing ice from the ice baskets identified by NRC inspectors were readily apparent conditions not previously identified by IMP staff. Further, NRC intervention was necessary to prompt licensee corrective actions for numerous deficiencies associated with the ice condenser such as missing or broken ice basket sheet metal screws found repeatedly by IMP staff in the ice melt system since 1991 without investigation or corrective action. The failure to effectively implement the corrective action program represented a programmatic breakdown in the control of licensed activities such that conditions adverse to quality were not aggressively pursued and resolved.

The sixteen violations in section C represent a programmatic breakdown of IMP’s design change program. Design control deficiencies resulted in the degraded condition of the ice condenser, containment sump, and the RWST level instruments. For the ice condenser, IMP failed to follow the design control process pertaining to changes in the method to secure ice baskets in place, and the repair of damaged baskets. For the containment sump, IMP failed to implement adequate controls for the installation of material in the containment that would have affected long-term post-LOCA recirculation cooling. Most notable was the routine installation of fibrous insulation material without appropriate controls. For the RWST, IMP failed to verify the adequacy of instrument uncertainty calculations which allowed the establishment of improper swap over setpoints. This condition could result in insufficient water inventory in the containment sump for ECCS during a LOCA also resulting in reduced/inadequate core cooling.

The seven violations in section D represent a programmatic breakdown of IMP's ability to perform safety evaluations to adequately assess the consequences of changes and ensure the plant was maintained as designed and specified in the licensing basis. For example IMP created an unreviewed safety question and a single failure vulnerability when they changed the proceduralized system lineup to transfer ECCS pump suction from the RWST to the containment sump using the west residual heat removal (RHR) pump. Specifically, failure of this RHR pump would cause the loss of both trains of emergency core cooling. Another example included several safety evaluations that failed to identify that operating the facility with the ultimate heat sink above its maximum temperature was an unreviewed safety question. Operation under these conditions could have affected the ability to reach cold shutdown. In addition, when the licensee did address elevated equipment operating temperatures, the associated safety evaluation failed to provide the basis for the determination that the higher temperatures were not an unreviewed safety question.

The violations in the four sections of the Notice have been collectively categorized in accordance with the NRC Enforcement Policy (NUREG-1600) as a Severity Level II problem. This Severity Level is warranted for the breadth and number of the violations that, taken in total, resulted in a lack of reasonable assurance that following a design basis LOCA, i.e., large break, the ECCS and containment would have functioned.

Accordingly, I have been authorized, after consultation with the Commission to exercise discretion pursuant to Section VII.A.1 of the NRC Enforcement Policy to assess a penalty in the amount of $500,000. Specifically, the escalated civil penalty reflects the consideration of the particularly poor licensee performance, the duration of the problems, the impact on ECCS and containment, and the NRC's concerns regarding the violations. The purpose of this enforcement action is to emphasize: (1) the need to take timely and effective corrective actions for identified deficiencies, (2) the need for effective surveillance testing and for plant personnel to challenge and investigate discrepancies identified during surveillance activities, (3) the need for rigorous safety evaluations to determine if changes to the plant or procedures constitute unreviewed safety questions, (4) the need to maintain systems' design bases, and (5) the need for a strong self-assessment program. The staff would have proposed higher civil penalty had it not been for IMP's decision to take comprehensive corrective actions and commitment to keep the facility shutdown until these problems are resolved.

Finally, the violations described in the Notice are not all of the apparent violations present or identified during the various inspections, but serve to represent the systemic nature of the significant regulatory problems existing at the D.C. Cook facility. The breadth and number of violations identified resulting in the significant degradation of multiple systems raise questions about the condition of other safety systems at D.C. Cook. This enforcement action emphasizes the need for IMP's ongoing review of the condition of other systems to be effective. Other apparent violations described in the inspection reports referenced in the Notice are not being addressed in this enforcement action. Nevertheless, they need to be considered as part of your corrective actions.

IMP is required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing its response. The NRC will use IMP's response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and IMP's response will be placed in the NRC Public Document Room (PDR). IMP's response may, as appropriate, make reference to the material IMP provided at the predecisional enforcement conference on May 20, 1998. To the extent possible, IMP's response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction.

Sincerely,

Original Signed By

James L. Caldwell
Acting Regional Administrator

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

Enclosure: Notice of Violation and
Proposed Imposition of Civil Penalty

cc w/encl: J. Sampson, Site Vice President
R. Eckstein, Chief Nuclear Engineer
D. Cooper, Plant Manager
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Indiana Michigan Power Company
Donald C. Cook Nuclear Plant

Docket Nos. 50-315; 50-316
License Nos. DPR-58, DPR-74
EA Nos. 98-150, 98-151,
98-152 and 98-186

During NRC inspections conducted from August 4, 1997 through April 15, 1998, violations of NRC requirements were identified. In accordance with NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," the NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalty are set forth below:

A. Performance of Inspection and Test Activities for Continued Availability and Operability of Safety Systems

1. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions and procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions and procedures.

   a. Contrary to the above, as of February 27, 1998, the licensee had not provided instructions appropriate to the circumstances for an activity affecting quality in that visual examinations of ice condenser flow passages using procedure 12 EHP 4030 STP.250 (Revision 1), "Inspection of Ice Condenser Flow Passages," failed to detect ice blockages in the flow passages. Specifically, this procedure lacked instructions to perform visual examinations from accessible areas above and below the ice condenser flow passages. Further, the procedure permitted an arbitrary flow passage selection process to be used by the Test Engineer which resulted in non representative samples being examined. (01012)

   b. Contrary to the above, as of February 27, 1998, the licensee failed to ensure that instructions appropriate to the circumstances for an activity affecting quality were provided in procedure 12 EHP 4030 STP.211 (Revision 2), "Ice Condenser Surveillance." Specifically, step 4.8 of procedure 12 EHP 4030 STP.211 authorized unpinning up to 60 ice baskets in Modes 3 and 4 without an analysis to determine if the integrity of the containment structure was affected with the ice condenser in this condition. (01022)

c. Procedure No. 01-OHP 4030.001.002 (Revision 14), "Containment Inspection Tours," defines how to perform containment inspections, an activity affecting quality.

Contrary to the above, as of September 11, 1997, 01-OHP 4030.001.002 was not appropriate to the circumstances because it did not require an individual to look for insulation that could restrict flow to the containment recirculation sump. Specifically, Fiberfrax insulation material was installed in 1985, 1986, and in 1995 during maintenance outages, and Temp-mat insulation was installed in 1989. Numerous containment inspections were made by the licensee during the last 12 years which never identified the need to remove fibrous insulation material. On September 11, 1997, fibrous insulation material which could restrict flow to the containment recirculation sump was found installed in the Unit 2 containment. (01032)

2. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

a. Contrary to the above, as of February 27, 1998, the licensee had failed to adequately incorporate the acceptance limit of design document WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," dated October 1988 into test procedure 12 EHP 4030 STP.250, "Inspection of Ice Condenser Flow Passages." Specifically, test procedure 12 EHP 4030 STP.250 incorporated the 15 percent uniform ice condenser flow blockage acceptance criterion of WCAP-11902 without accounting for measurement errors, which when considered in the procedure, would result in a flow passage blockage acceptance criterion in excess of that previously analyzed. (01042)

b. Contrary to the above, as of February 27, 1998, the licensee had failed to adequately incorporate the analyzed acceptance limit (Westinghouse evaluation "Indiana Michigan Power D.C. Cook Nuclear Power Plant Ice Condenser Seismic Load Study New Ice Basket Design," dated February 28, 1990) for the combined ice basket with ice weight (gross ice basket weight) into Attachment 4, "Ice Condenser Basket Work Sheet," of test procedure EHP 4030 STP.211, "Ice Condenser Surveillance," Revision 2. Specifically, the 1877 lb. acceptance criterion used in the procedure did not account for measurement errors, which when considered, would result in a maximum gross ice basket weight acceptance criterion in excess of that previously analyzed. (01052)

3. 10 CFR Part 50, Appendix B, Criterion VII, “Control of Purchased Material, Equipment, and Services,” requires, in part, that the effectiveness of the control of quality by contractors shall be assessed at intervals consistent with the importance, complexity and quantity of services.

   Contrary to the above, the licensee had not assessed the effectiveness of the control of quality by the ice basket weighing contractors performing ice condenser surveillance testing since the 1995 refueling outage. Numerous ice baskets sustained potentially detrimental damage. Specifically, on March 3, 1998, November 12, 1997, and February 28, 1997, the licensee attributed ice baskets damage (documented in CR 98-388, CR 97-3244, CR 97-0544) to weighing practices and associated activities performed by contractors during ice condenser surveillance testing. (01062)

4. Technical Specification 4.6.5.1.d, “Ice Condenser - Ice Beds,” requires, in part, that the licensee visually inspect accessible portions of at least two ice baskets from each 1/3 of the ice condenser and verify that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage.

   Contrary to the above, on March 20, 1997, the licensee visually inspected the accessible portion of ice basket 6-3-4 (a basket selected for the Technical Specification 4.6.5.1.d inspection) but failed to verify the basket was free of detrimental structural wear, cracks, corrosion or other damage in the applicable surveillance procedure 12 EHP 4030 STP.212 (Revision 0) “Ice Condenser Basket Inspection.” Specifically, the licensee failed to identify structural damage at ice basket 6-3-4 lower rim assembly which was accessible. (01072)

5. Technical Specification Surveillance Requirement 4.6.5.1.b.2 requires, in part, that the licensee weigh a representative sample of at least 144 ice baskets and verify that each ice basket contains at least 1333 pounds of ice.

   Contrary to the above, during the 1995 refueling outage, the licensee failed to select a representative sample of ice baskets to meet Technical Specification 4.6.5.1.b.2 for the ice weight surveillance. The selected ice baskets constituted a non-representative sample, in that azimuthal row 5 ice baskets were excluded, which were lighter than other azimuthal rows (e.g., contained a significant percentage of ice baskets below the 1333 pounds of ice required). Further, the selection was nonrepresentative in that the same ice baskets were repetitively weighed (particularly in radial rows 8 and 9) during sequential surveillance intervals. (01082)

B. Implementation of a Corrective Action Program to Assure Conditions Adverse to Quality are Effectively Corrected

10 CFR Part 50, Appendix B, Criterion XVI requires, in part, that measures shall be established to ensure that conditions adverse to quality such as defective material and non conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

1. Contrary to the above, as of January 25, 1998, the licensee failed to identify or implement corrective action for the failed ice basket sheet metal screws, a condition adverse to quality, which had been repeatedly found in the ice melt system filters for both units since 1991. (01092)

2. Contrary to the above, as of February 4, 1998, the licensee failed to identify, or implement corrective action for the numerous ice baskets in Units 1 and 2 with missing ice segments (six to eighteen feet in length) representing a significant reduction of basket ice mass, which was a condition adverse to quality, located near the lower end of the ice basket. (01102)

3. Contrary to the above, as of February 4, 1998, the licensee failed to identify or implement corrective actions for the dented/buckled webbing, a condition adverse to quality, located near the bottom ice basket rim assembly on more than 40 Unit 1 and more than 100 Unit 2 ice baskets. (01112)

4. Contrary to the above, as of February 27, 1998, the licensee failed to implement adequate measures to preclude repetition of loose U-bolt nuts at the bottom ice basket assembly, significant conditions adverse to quality. Loose U-bolt nuts were identified on ice baskets in 1990 for Unit 1 (documented in PR 90-1639). Preventive actions taken by the licensee to preclude recurrence of this condition included modifying surveillance procedure 12 THP 4030 STP.211, “Ice Condenser Surveillance” to inspect ice baskets for loose or missing nuts. Subsequently, loose U-bolts were again identified on Unit 1 ice baskets in 1992 (documented in PR 92-1386) and in Unit 2 (documented in PR 92-0360). (01122)

5. Contrary to the above, as of February 27, 1998, the licensee failed to implement adequate measures to identify the cause and preclude repetition of separated Unit 1 ice basket assemblies, a significant condition adverse to quality. The licensee had not established a definitive root cause for the separated ice baskets documented in CR 1-07-83-647 and CR 1-08-83-771. Further, no corrective action measures had been implemented for these failures. On February 28, 1997, the licensee identified another separated basket as documented in

CR 97-0554. Again, the licensee failed to determine the cause for the separated basket and did not implement any corrective actions to preclude recurrence. (01132)

6. Contrary to the above, as of February 27, 1998, the licensee failed to implement adequate measures to identify the cause and preclude repetition of failed fillet welds at the ice basket bottom hold down bar, a significant condition adverse to quality. Licensee corrective actions completed in 1992 and documented in PR 92-1181 for failed fillet welds at the ice basket bottom hold down bar were not adequate to resolve this significant condition adverse to quality. Specifically, FSAR Appendix M, Section 3.1.4 required application of the design basis accident loads in qualifying the design of the ice baskets. WCAP-8304, "Stress and Structural Analysis and Testing of Ice Baskets," dated May 1974, defined the design basis accident lateral and compressive loadings used in analysis and testing of the original ice baskets. Licensee engineering evaluations dated July 27 and August 13, 1992, failed to apply these lateral or compressive loadings in accepting the ice baskets with the failed fillet welds. (01142)

C. Control and Maintenance of the Facility Design Basis

1. 10 CFR 50.9(a) requires, in part, that information required by statute or by the Commission's regulations, order, or license condition to be maintained by the licensee shall be complete and accurate in all material respects.

   10 CFR 50.71(e), "Maintenance of Records, Making of Reports," requires, in part, that each person licensed to operate a nuclear power reactor shall update periodically, the final safety analysis report (FSAR) to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirements since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of all changes made in the facility or procedures as described in the FSAR and all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question.

   a. Contrary to the above, as of February 27, 1997, the licensee failed to update FSAR Section 5.3.1, "Design Consideration," to incorporate analysis WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," dated October 1988, which established the limit for ice condenser flow passage blockages used as the basis for the acceptance criterion in surveillance procedure 12 EHP 4030 STP 250 (Revision 1), "Inspection of Ice Condenser Flow Passages." WCAP-11902 is a safety evaluation that was submitted to the Commission in support of a license amendment request for new limits for an ice condenser flow passage blockage. The information the licensee submitted to the NRC in the FSAR was not complete and accurate. (01152)

b. Contrary to the above, as of February 27, 1998, the licensee failed to update FSAR Figure 6.4.1, "Typical Bottom Ice Basket Assembly," of FSAR Appendix M, "Ice Condenser Component Evaluation Report," to conform to the as-built ice basket bottom assembly configuration that involves a welded hold down bar, versus a bolted rectangular tube support assembly. The information the licensee submitted to the NRC in the FSAR was not complete and accurate. (01162)

c. Contrary to the above, as of February 27, 1998, the licensee failed to update FSAR, Appendix M, Section 6.4.2 to incorporate the latest material developed. Specifically, the following modifications made to the facility as described in the FSAR had not been included in a licensee update submittal. The information the licensee submitted to the NRC in the FSAR was not complete and accurate. (01172)

   i. Modification 02-MM-032, "Ice Basket Reinforcement - Problem Report #88-914," installed clamps, a pipe brace, and a cable to repair a damaged Unit 2 ice basket on February 10, 1989.

   ii. Modification 01-MM-048, "Minor Modification Temporary Repair of Damaged Ice Baskets," installed clamps, a pipe brace and cables to repair eight damaged Unit 1 ice baskets on July 11, 1989.

d. Contrary to the above, as of February 27, 1998, the licensee failed to update FSAR, Appendix M, Table 4.3 -1 to incorporate the current maximum analyzed ice basket weight of 1877 lbs., which had been established in a Westinghouse evaluation "Indiana Michigan Power D.C. Cook Nuclear Power Plant Ice Condenser Seismic Load Study New Ice Basket Design" dated February 28, 1990, accepted by the licensee on March 1, 1990, and incorporated into surveillance procedures. The information the licensee submitted to the NRC in the FSAR was not complete and accurate. (01182)

2. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that the licensee shall establish measures to assure that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design. Further, these measures shall assure that the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions; and that design

control measures provide for verifying or checking the adequacy of design, such as the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

*[1]a. Contrary to the above, as of February 19, 1998, changes had been made to Unit 1 ice baskets without being subject to design control measures commensurate with those applied to the original design. Specifically, a galvanized bolt had been installed in place of the clevis pin that connected the ice basket to the support structure for ice baskets 4-1-9, 5-9-1 and 20-3-6. (01192)

*b. Contrary to the above, as of February 19,1998, changes had been made to a Unit 2 ice basket without being subject to design control measures commensurate with those applied to the original design. Specifically, a six-inch wide curved sheath of sheet metal had been installed onto the ice basket mesh of ice basket 1-7-9. (01202)

*c. Contrary to the above, as of February 19, 1998, changes had been made to a Unit 2 ice basket without being subject to design control measures commensurate with those applied to the original design. Specifically, nine rivets had been installed in place of sheet metal screws at the bottom ice basket rim coupling of ice basket 14-6-8. (01212)

d. Contrary to the above, inadequate measures were established to assure that the containment sump design basis was correctly translated into specifications for the installation of Fiberfrax refractory insulation in the containment. Specifically, FSAR Section 6.2.2, “ECCS, System Design and Operation,” states, in part, that the containment sump provided adequate net positive suction head for the residual heat removal pumps and containment spray pumps to operate in the recirculation mode. However, specification DCC-FP101-QCN (Revision 14 and Change Sheet 1), dated February 28, 1995, “Fire Barrier Penetration Seals,” Section 3.5, which details the requirements for the installation and maintenance of fire barrier penetration seals and fire stops states that Fiberfrax refractory insulation can be left in place in containment following the sealing operation. Further, procedure no. 12CHP5021.ECD.005 (Revision 9), “Installation, Replacement, and Repair of Silicone Fire

[1]Violations annotated with an asterisk (*) are violations which occurred beyond the five year statute of limitations period for assessing civil penalty or are violations for which definitive dates to establish their occurrence are unavailable to determine the statute of limitations applicability but likely occurred more than five years before the inspection. In either case, these violations were not considered for purposes of determining any civil penalty.

Barrier Penetration Seals," which provides the instructions for cable tray and conduit fire barriers/stops installation, permitted Fiberfrax damming material to remain in place following the installation in containment. Following a LOCA, Fiberfrax material could become dislodged and collect on the sump suction strainers restricting post loss of coolant accident recirculation capability. (01222)

*e. Contrary to the above, inadequate measures were established to assure that the containment sump design basis was maintained and correctly translated into specifications because the specifications were changed without using design control measures commensurate with those applied to the original design. Specifically, the Updated Final Safety Analysis Report at Section 6.2.2, "ECCS, System Design and Operation," states, in part, that the containment sump provided adequate net positive suction head for the residual heat removal and containment spray to operate in the recirculation mode. Specification DCC-PV450-QCS (Revision 6), "Thermal Insulation," at Section 4.3.9, "Metal Jackets Within Containment," states, in part, that all applied pipe insulation within the containment area shall be covered with prefabricated 0.010" thick, type 304 stainless steel jackets. However, a January 25, 1989 memorandum permitted the use of Temp-mat insulation without a 0.010" thick stainless steel (type 304) jacket as a replacement for metallic insulation contrary to Design Specification DCC-PV450-QCS, and incorrectly indicated that "the replacement is not considered to be a design change." This design change was not subject to design control measures that were commensurate with the original design. Following a LOCA, without the metal jackets, Temp-mat debris could be swept from its installed location and be transported to the containment sump where it would block the sump screens and contribute to degraded post loss of coolant accident recirculation capability. (01232)

f. Contrary to the above, the licensee failed to ensure that the RWST design basis was correctly translated into specifications by failing to implement measures to verify or check the adequacy of instrument uncertainty calculation Engineered Control Procedure (ECP) 1-RPC-09 (Revision 2), "Refueling Water Storage Tank (RWST) Level" dated December 2, 1993. Specifically, the RWST level channel uncertainty calculation did not include the RWST discharge pipe entrance friction head loss and the velocity head loss during maximum emergency core cooling flow rates. These head losses (biases) caused the indicated RWST level to read lower than actual tank level. This could affect emergency core cooling system (ECCS) and containment spray (CTS) pumps suction transfers from the RWST to the containment recirculation sump during a design basis accident. The premature transfer could cause ECCS and CTS pump loss due to vortexing (air entrainment) and/or the loss of net positive suction head (NPSH) from insufficient sump water level. (01242)

g. Contrary to the above, the licensee failed to ensure that the RWST design basis was correctly translated into specifications by failing to implement measures to verify or check the adequacy of instrument uncertainty calculation No. ECP 1-CG-39 (Revision 1), “Refueling Water Storage Tank (RWST) Level” dated October 21, 1994. Specifically, the RWST level channel uncertainty calculation did not include vortexing or air entrainment that could occur at the RWST discharge pipe during maximum emergency core cooling flow rates before the suction for the pumps was transferred from the RWST to the containment sump. Vortexing could cause ECCS and CTS pump loss due to air binding. (01252)

h. Contrary to the above, the licensee failed to ensure that the ECCS design basis was correctly translated into specifications by failing to implement measures to verify or check the adequacy of instrument uncertainty calculations ECP 1-2-N3-01, 1-RPC-14, and 2-RPC-14, Revisions dated March 16, 1994, May 17, 1994, and May 17, 1994, respectively. Specifically, the containment sump level instrumentation loops did not account for the loop uncertainty impact on post-accident containment levels, did not include considerations for residual heat removal (RHR) and CTS pumps NPSH requirements, and did not account for pump vortexing (air entrainment). As a consequence, this could impact ECCS or CTS pumps during transfer from the RWST to the containment sump when implementing emergency operating procedure 01(02)-OHP 4023.ES-1.3, “Transfer to Cold Leg Recirculation.” (01262)

*i. Contrary to the above, as of September 10, 1997, the licensee did not correctly translate the required containment water inventory design into specifications, drawings, procedures, and instructions. Specifically, engineering reviews did not evaluate the effects of reactor coolant flow diversions into the inactive portions of the containment sump where it would not be available during a design basis accident. Therefore, it was not known if sufficient water could be recovered during a design basis accident to prevent ECCS or containment spray pump vortexing (air entrainment) during containment sump recirculation. This could jeopardize long term pump operation. (01272)

j. Contrary to the above, in 1996 and 1997, the licensee failed to translate into specifications, drawings, procedures, and instructions for the design basis of the ¾-inch containment recirculation sump roof vent hole. The design basis, which was to minimize air entrapment under the containment sumps roof slab, was specified in AEP:NRC:00110, dated December 29, 1978. However, in 1996 for Unit 2 and 1997 for Unit 1, the licensee sealed the vent holes without using the design control process. (01282)

*k. Contrary to the above, the licensee did not correctly translate the ¼-inch containment recirculation sump particulate retention design basis into specifications, drawings, procedures, and instructions. Specifically, design change DC-12-236, dated March 27, 1979, was deficient because it permitted the installation of fine particulate screens with gaps in excess of ½-inch at the edges of individual screen sections together with no screens over the ¾-inch sump vent holes. As a consequence, a common mode failure of both CTS trains could have occurred because of the size of the particles that was permitted to enter the sump. The screens' purpose was to prevent introduction of debris that could plug the ⅜-inch containment spray nozzles. (01292)

*l. Contrary to the above, the licensee had not implemented adequate measures to assure that the correct design values were used to calculate the maximum heat loading for the containment spray heat exchanger room per DCCHV12AE06N, dated June 3, 1992, "Heat Gain Calculation - AES System." Specifically, the calculation incorrectly used an essential service water flow of 3300 gpm and a containment sump inlet temperature of 170°F. According to FSAR Table 9.8-5, "Essential Service Water System Minimum Flow Requirement," at note 4 the minimum essential service water flow was 2400 gpm and according to FSAR section 6.3.2, "System Design," the maximum containment sump temperature was 190°F. (01302)

D. Conduct of Safety Evaluations to Assure Facility and Procedure Changes do not Create Unreviewed Safety Questions.

1. 10 CFR 50.59(a)(1), "Changes, Tests and Experiments," states, in part, that the holder of a license authorizing operation of a utilization facility may, (1) make changes in the facility as described in the safety analysis report, and (2) make changes in the procedures as described in the safety analysis report without prior Commission approval, unless the proposed change involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

   a. 10 CFR 50.59 (a)(2) states, in part, that a proposed change, test, or experiment shall be deemed to involve an unreviewed safety question:

(1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) if the margin of safety as defined in the basis for any technical specification is reduced.

i. Contrary to the above, safety evaluations of March 11and March 20, 1996, for the core off-load were inadequate because they failed to recognize that the Unit 1 CCW system could not perform its function under the design basis assumptions described in the FSAR and failed to conclude that this change involved an unreviewed safety question. Specifically, during the Unit 2 full core off-load outage and with Unit 1 at 100% power, both Unit 2 CCW and essential service water (ESW) trains were taken out-of-service, leaving one Unit 1 CCW train available to supply spent fuel pool cooling. A single CCW train operating at 95°F could not maintain the spent fuel pool (SFP) bulk water temperature less than the temperature (160°F) specified in FSAR Section 9.4, “Spent Fuel Pool Cooling System.” In addition, with a single Unit 1 CCW train providing SFP cooling, a Unit 1 design basis accident would isolate CCW causing a loss of SFP cooling. As a consequence, the SFP time-to-boil margin could be reduced to less than the 5.74 hours specified in FSAR. Operation of the facility with one unit off loaded, the other unit at full power operation, and only one train of spent fuel pool cooling available created the possibility for an accident or malfunction of a type not previously evaluated in the FSAR. (01312)

ii. Contrary to the above, during July and August of every year between 1994 and 1997, the licensee made a change to the facility without Commission approval, that involved an unreviewed safety question (USQ). Specifically, the licensee made a change by operating the facility above its maximum ultimate heat sink (lake) temperature limit (76°F) as stated in FSAR Tables 6.3-2 and 9.5-3. However, no safety evaluation was performed and the UFSAR had not been updated to reflect operation above the 76°F limit. For example, on July 17, July 18 and August 4 of 1997, the temperature exceeded the 76°F limit. Operating the facility with the ultimate heat sink above its maximum temperature involved a USQ because the higher temperatures increased the probability for failure of equipment important to safety previously evaluated in the UFSAR. (01322)

2. 10 CFR 50.59, "Changes, tests and experiments," in part, permits the licensee to make changes to its facility and procedures as described in the safety analysis report and conduct tests or experiments not described in the safety analysis report without prior Commission approval provided the change does not involve a change in the technical specifications or an Unreviewed Safety Question (USQ). The licensee shall maintain records of changes in the facility and these records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

   10 CFR 50.71(e) requires, in part, a licensee to update the FSAR originally submitted as part of the application for the operating license to assure that the information included in the FSAR contains the latest material developed. The updated FSAR shall be revised to include the effects of, in part, all safety evaluations performed by the licensee in support of conclusions that changes did not involve a USQ.

   10 CFR 50.9(a) requires, in part, that information provided to the NRC by a licensee or information required to be maintained by a licensee shall be complete and accurate in all material respects.

   1. Contrary to the above, from June 1992 to January 1997, the facility was not in conformance with the FSAR in that the licensee revised emergency operating procedure Nos. 01(02) - OHP 4023.ES -1.3, Revision 2, “Transfer to Cold Leg Recirculation,” to operate in series (piggy-back) both centrifugal charging and safety injection trains onto the west residual heat removal (RHR) pump and there was not an adequate safety evaluation performed to determine that there was not a unreviewed safety question. Specifically, FSAR Section 6.2.2 stated that the transfer to cold leg recirculation is performed by trains and specified a transfer sequence from the injection phase to the recirculation phase. However, because the west RHR pump would be operating to supply both centrifugal charging and safety injection pumps, the failure of the west RHR pump would cause the loss of all emergency core cooling. In addition, ES-1.3, Revision Nos. 3 and 4, and their corresponding safety evaluations failed to identify the single failure vulnerability and the fact that the FSAR section 6.2.2 specified a transfer sequence from the injection phase to the recirculation phase that was not implemented by ES-1.3. As a result, the 50.59 safety evaluation for this procedure revision failed to identify that an unreviewed safety question (single failure vulnerability) was created by this procedure change because of the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report was created. In addition the updated FSAR was not complete and accurate in all material

respects in that it did not reflect this change in operation of the plant. (01332)

ii. Contrary to the above, as of September 10, 1997, the licensee had operated the component cooling water (CCW) system at temperatures (120°F) above FSAR Table 9.5.3 specified design value of 95°F without a written safety evaluation providing the basis for the determination that operating the reactor coolant pump (RCP) seals with higher CCW temperatures was not an unreviewed safety question. In addition the updated FSAR was not complete and accurate in all material respects in that it did not reflect this change in operation of the plant. (01342)

iii. Contrary to the above, as of September 10, 1997, the licensee operated the RCP thermal barrier heat exchanger, for both units, with a CCW flow between 25 and 35 gpm for a total flow of 100 - 140 gpm without a written safety evaluation providing the bases for the determination that operating with reduced RCP thermal barrier heat exchanger flow was not an unreviewed safety question. Specifically, FSAR Table 9.5-2 stated that the minimum flow was 140 gpm total or a minimum flow of 35 gpm to each RCP thermal barrier. However, the licensee operated the RCP thermal barriers with flow as low as 25 gpm. In addition the updated FSAR was not complete and accurate in all material respects in that it did not reflect this change in operation of the plant. (01352)

c. 10 CFR 50.59 (b)(1) requires, in part, that the licensee shall maintain records of changes in the facility made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the safety analysis report. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

i. FSAR Table 9.5.2 “Component Cooling Water System Minimum Flow Requirements Per Train (GPM)” listed the letdown heat exchanger <u>maximum</u> flowrate during normal and cooldown operations as 984 gpm.

Contrary to the above, safety evaluation SECL-97-198, ”FSAR Change to Support Increased CCW Temperature,” dated November 12, 1997, was inadequate in that an evaluation had not been performed to determine that the change to the system configuration specified in FSAR Table 9.5.2 did not involve an unreviewed safety question. Specifically, the letdown heat exchanger control system could automatically open the CCW outlet flow control valve in an attempt to maintain outlet temperature at 120°F causing flow to potentially reach 1400 gpm. No written evaluation was performed to address this change from the FSAR design maximum flow of 984 gpm. (01362)

ii. FSAR Section 6.2.2, “System Design and Operation,” page 6.2-12, describes the changeover from the injection phase to the recirculation system phase. Specifically, this section describes the low level setpoint of the refueling water storage tank as 131,980 gallons.

Contrary to the above, procedure no. 01(02)-OHP 4023.ECA-0.2 allowed plant operation with the low level setpoint changed from 31 percent to 20 percent. The 10 CFR 50.59 screening, dated January 3, 1998, evaluating the change, failed to recognize and evaluate the change to the plant as described in FSAR Section 6.2, which listed a volume of 131,980 gallons which corresponds to 31 percent of the tank volume for the low level setpoint. (01372)

These violations represent a Severity Level II problem. (Supplement I) – Civil Penalty $500,000

Pursuant to the provisions of 10 CFR 2.201, Indiana Michigan Power Company (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted, and if denied, the reasons why; (3) the corrective steps that have been taken and the results achieved; (4) the corrective steps that will be taken to avoid further violations; and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalty proposed above in accordance with NUREG/BR-0254 and by submitting, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalty will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation(s) listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.B.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim for withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 13th day of October 1998