DONALD C. COOK NUCLEAR PLANT
1992 ANNUAL OPERATING REPORT

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1.0 INTRODUCTION

1.1 PLANT DESCRIPTION

The Donald C. Cook Nuclear Plant is owned by Indiana Michigan Power Company and is located five miles north of Bridgman, Michigan. The plant consists of two nuclear power units, each employing a Westinghouse pressurized water reactor nuclear steam supply system. Each reactor unit employs an ice condenser reactor containment system. The American Electric Power Service Corporation was the architect-engineer and constructor.

Unit 1 and 2 reactor design power output (and licensed rating) are 3250 MWt and 3411 MWt, respectively. Unit 1 approximate gross and net electrical outputs are 1056 MWe and 1020 MWe, respectively. Unit 2 approximate gross and net electrical outputs are 1100 MWe and 1060 MWe, respectively. The main condenser cooling method is open cycle using Lake Michigan water as the cooling source for each unit.

1.2 REPORT PREPARATION

This report was compiled by W. R. Moran with the following individuals contributing information as follows:

D. L. Noble	- Personnel Exposure Summary
C. A. Freer	- Steam Generator ISI Summary
R. G. Vasey	- Changes to Procedures
B. A. Svensson	- Challenges to Pressurizer PORVs and Safety Valves
R. G. Vasey	- Tests or Experiments Not Described in the FSAR
B. A. Svensson	- Reactor Coolant Specific Activity
T. A. Georgantis	- Results of Irradiated Fuel Inspections
R. G. Vasey	- Changes to Facility - RFCs, MMs, PMs
R. G. Vasey	- Changes to Facility - Temporary Modifications to Unit 1 & 2

2.0 PERSONNEL RADIATION EXPOSURE SUMMARY

Table 1 provides a summary of the number of station, utility, and contractor (and others) personnel receiving exposures greater than 100 millirem in 1992. The total record dose for all personnel was 492.044 rem as measured by thermoluminescent dosimetry (TLD) and reported in accordance with Regulatory Guide 1.16.

TABLE 1

ANNUAL OPERATING REPORT - RG 1.16 FOR 1992

	# PERSONNEL >100 mR			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
Reactor Operations & Surveillance						
Maintenance Personnel	0002	0000	0008	000.477	000.000	001.262
Operations Personnel	0046	0002	0006	010.275	000.580	001.163
Health Physics Personnel	0031	0000	0070	007.956	000.000	024.259
Supervisory Personnel	0001	0000	0000	000.202	000.000	000.000
Engineering Personnel	0000	0000	0002	000.000	000.000	000.325
Routine Maintenance						
Maintenance Personnel	0083	0001	0298	024.006	000.238	118.059
Operations Personnel	0009	0001	0054	002.483	000.118	029.311
Health Physics Personnel	0009	0000	0032	001.730	000.000	005.886
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0009	0000	0002	002.054	000.000	000.744
In-Service Inspection						
Maintenance Personnel	0003	0000	0076	000.377	000.000	027.172
Operations Personnel	0003	0000	0019	000.632	000.000	006.341
Health Physics Personnel	0001	0000	0016	000.131	000.000	003.326
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0001	0000	0006	000.117	000.000	001.890
Special Maintenance						
Maintenance Personnel	0013	0000	0166	002.124	000.000	051.826
Operations Personnel	0000	0000	0010	000.000	000.000	002.897
Health Physics Personnel	0002	0000	0018	000.217	000.000	002.691
Supervisory Personnel	0001	0000	0000	000.112	000.000	000.000
Engineering Personnel	0001	0001	0007	000.102	000.367	002.616
Waste Processing						
Maintenance Personnel	0000	0000	0003	000.000	000.000	001.296
Operations Personnel	0000	0000	0004	000.000	000.000	001.162
Health Physics Personnel	0004	0000	0066	000.600	000.000	012.730
Supervisory Personnel	0001	0000	0000	000.206	000.000	000.000
Engineering Personnel	0000	0000	0000	000.000	000.000	000.000
Refueling						
Maintenance Personnel	0004	0000	0031	000.558	000.000	006.158
Operations Personnel	0006	0000	0038	001.955	000.000	013.789
Health Physics Personnel	0000	0000	0032	000.000	000.000	005.049
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0002	0000	0001	000.495	000.000	000.158
TOTALS						
Maintenance Personnel	0089	0001	0461	027.542	000.238	205.773
Operations Personnel	0059	0003	0104	015.345	000.698	054.663
Health Physics Personnel	0032	0000	0150	010.634	000.000	053.941
Supervisory Personnel	0002	0000	0000	000.520	000.000	000.000
Engineering Personnel	0010	0001	0016	002.768	000.367	005.733
GRAND TOTALS	0192	0005	0731	056.809	001.303	320.110

3.0 STEAM GENERATOR IN-SERVICE INSPECTION

3.1 UNIT 1 INSPECTION SUMMARY (SEE ENCLOSURE)

The Donald C. Cook Unit 1 steam generator inspection/repair program performed during July and August of 1992 was an extensive program. In addition to conducting the basic scope of bobbin coil (BC) and rotating pancake coil (RPC) inspections of inservice tubes, 364 tubes potentially defective alloy 600 plugs, susceptible to primary water stress corrosion cracking were removed from their hot leg tubes (see enclosure). A selected number of these tubes were also inspected with a BC probe since they were considered repairable by sleeving. This resulted in 209 of these tubes, among four steam generators, being returned to service with sleeves. The scope of these efforts, combined with removing samples from four different tubes within steam generator 12 to be destructively examined, are listed in Sections I through III of the enclosure to this report.

3.2 UNIT 2 INSPECTION SUMMARY

Eddy current inspection of Unit 2 generators was performed from March 19 to March 28, 1992. Approximately 6.5% of the total number of tubes in SGs 21 and 24 received an eddy current bobbin coil inspection to the extent as shown in Table 2.

TABLE 2
UNIT 2 STEAM GENERATOR INSPECTION

	S/G 21	S/G 24
Inspected full length from hot leg	71	69
Inspected to 7C from hot leg*	159	163
Inspected to 6C from hot leg*	4	2
Inspected to 5C from hot leg*	1	1

No imperfections were found and no plugs were installed. Both of these SGs were returned to service as found.

* 5C, 6C, and 7C indicates that the length of the inspection from the hot leg side was to the fifth, sixth, and seventh support plate on the cold leg side.

4.0 CHANGES TO PROCEDURES

This section contains a brief description of the procedure changes implemented under the provisions of 10 CFR50.59 and the associated safety evaluations.

4.1 MAINTENANCE PROCEDURES

4.1.1 Fuse Testing

Description of Change:

Plant procedure 12 MHP 5021.082.016, which required periodic sample testing of standard grade fuses installed in circuits fed by battery-backed power buses, was canceled. This procedure is no longer necessary because safety grade fuses are now installed.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the procedure was initially instituted as a test program designed to provide assurance that safety related fuses, purchased as standard grade, would function properly. The fuses have been replaced with safety-grade fuses, providing assurance that the fuses will properly function.

4.2 OPERATIONS PROCEDURES

4.2.1 Operation of Containment Pressure Relief System

Description of Change:

Plant Procedure 02-OHP 4021.028.004, "Operation of the Containment Pressure Relief System," was changed to allow the use of the containment purge supply and exhaust system to relieve containment pressure. Because of this change, the exhaust would not pass through charcoal and HEPA filters as described in the FSAR.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the containment isolation valves would remain functional and containment activity levels would be checked prior to making any releases. Releases would only be made if the activity levels were acceptable.

4.2.2 Turbine Generator Testing

Description of Change:

Two procedures, 02-OHP SP.100 and -101, were written to conduct testing to determine the cause of turbine vibration that occurred in Unit 2. The testing consisted of increasing the turbine speed to 1930 rpm (rated speed is 1800 rpm), increasing the seal oil supply temperature to 122-131°F and increasing the number 5 and 6 bearing temperature limits to 285°F.

Safety Evaluation Summary

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the testing had been proposed and approved by the turbine manufacturer personnel who had concluded that it would not adversely impact the turbine integrity or result in an increased risk of turbine missile generation.

4.2.3 Transfer to Cold Leg Recirculation

Description of Change:

Plant procedures 01-,02-OHP 4023.ES-1.3, Revision 2 made changes in the valve manipulation procedure which differ from the description in Section 6.2 of the FSAR. In particular, the SI pump minimum flow valves are closed (Step 4 of the FSAR description) after the RHR and CTS pumps are started (Step 6 of the FSAR description).

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that these are emergency procedures, and the change in the valve manipulation sequence will not increase the consequences of an accident.

4.3 ENVIRONMENTAL MONITORING PROCEDURES

4.3.1 Spill Response Procedure Revisions

Description of Change:

Plant procedure PMI-2230, "Spill Response-Oil, Polluting, and Hazardous Materials," was revised to include additional items with regard to establishing an Incident Command System and Emergency Response Plan which meets Michigan Hazardous Waste Operations and Emergency Response requirements, and to address underground storage tank leaks. Four other procedures, PMP 2080 EPP.110, PMP 2081 EPP.103,.203, and .205 also required revisions as a result of these changes.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the changes to the procedure would not initiate an accident, would not adversely impact safety related equipment, and do not impact technical specification compliance.

5.0 TEST OR EXPERIMENTS NOT DESCRIBED IN THE FSAR

This section describes procedures classified as a "Test and Experiment" implemented under the provisions of 10 CFR50.59 including the associated safety evaluation.

5.1 TESTS

5.1.1 Turbine Generator Testing

See Section 4.2.2 - Turbine Generator Testing.

6.0 CHALLENGES TO PRESSURIZER POWER OPERATED RELIEF VALVES AND SAFETY VALVES

During 1992, there were no challenges on either Unit 1 or Unit 2 to the pressurizer power operated relief valves (PORV's) or the pressurizer safety valves as a result of the valves being called upon to mitigate an actual overpressure condition.

7.0 REACTOR COOLANT SPECIFIC ACTIVITY

During 1992, there were no instances on either Unit 1 or Unit 2 in which the reactor coolant I-131 specific activity exceeded the limits of Technical Specification 3.4.8.

8.0 IRRADIATED FUEL EXAMINATIONS

During 1992 three separate examinations were performed on the irradiated fuel discharged from Unit 2, Cycle 8, and two separate examinations were performed on the irradiated fuel discharged from Unit 1, Cycle 12. These examinations were conducted in parallel with, or shortly after, the core was unloaded. The intent was to determine fuel rod failures and gross structural defects in the fuel assemblies.

8.1 VISUAL EXAMINATIONS

The first examination of the irradiated fuel for each unit was by routine binocular inspections of the fuel assemblies per procedure number 12 SHP 4050 QC.002. As each assembly is off-loaded to the spent fuel pool, it is visually examined on all four sides. The examiner is looking specifically for torn or missing grid straps, missing or damaged fuel rods, excessive clad hydriding, or rod bow to gap closure. This inspection is primarily intended to detect fuel damage caused by mechanical interaction between fuel assemblies or baffle jetting, and is done during each refueling. There were no indications of any fuel damage for either unit.

8.2 ULTRASONIC EXAMINATIONS

The second examination of the irradiated fuel for each unit was performed by using ultrasonic testing (UT) methods. The ultrasonic system works by a probe transceiver sending a high frequency sound wave into a fuel rod and measuring the strength of the returning signal, or "ring back." A fuel rod can be determined to have water in it by monitoring the relative strength of this ring back. The presence of water in a fuel rod would indicate that the rod has a leak. In this way, not only can an assembly be determined to have leaking rods, but also the numbers and locations of the leaking rods can be identified.

For the Unit 2 UT inspection, the 117 irradiated fuel assemblies that were scheduled for reload into Cycle 9 were tested. Of these, two fuel assemblies were determined to each contain one leaking fuel rod. In order to identify suitable replacement fuel assemblies for the two that contained leaking fuel rods, an additional eight fuel assemblies were tested. The UT methods were employed. None of these additional eight fuel assemblies were found to contain leaking fuel rods.

For the Unit 1 UT inspection, the 113 irradiated fuel assemblies that were scheduled for reload into Cycle 13 were tested. None of the 113 fuel assemblies were found to contain leaking rods.

8.3 FUEL SIPPING EXAMINATION

The third examination of the irradiated fuel for Unit 2 was through use of the fuel sipping technique. The technique used for the Unit 2 fuel involved the placement of the fuel assembly to be tested in a

leak-tight canister, and a drawing of a partial vacuum in the canister. If the fuel assembly were to contain leaking fuel rods, radioactive gases would accumulate at the top of the canister. The gases at the top of the canister are measured for presence of radioactivity.

The fuel sipping results for the discharged fuel assemblies from Unit 2, Cycle 8 and the two fuel assemblies identified as replacements for the leaking fuel identified during UT confirmed the results of UT.

The Unit 2 fuel was sipped due to the results of the analysis of the reactor coolant system (RCS) radiochemistry data from Cycle 8. This data indicated the possibility of a "tight" leaking fuel rod, as the RCS radiochemistry data identified the presence of only gaseous fission products (e.g., xenon) and no indications of solid fission products (e.g., cesium). A concern existed that water may not have entered the "tight" leaking fuel rod and therefore, UT methods may have been inadequate to prevent reloading a leaking fuel rod. Therefore, it was decided to sip Unit 2 fuel.

9.0 CHANGES TO FACILITY

This section contains a brief description of the design changes implemented under the provisions of 10 CFR50.59 and the associated safety evaluations.

9.1 DESIGN CHANGES

9.1.1 Emergency Diesel Generator Control System Modification

Description of Change:

RFC-DC-12-2864 modified the emergency diesel generator control system to allow a controlled, slow start for surveillance testing, taking the diesels up to rated speed in an estimated 25 to 30 seconds. The modification included the installation of a timer to maintain the diesel generator in the slow start mode during the period that it is accelerating from 95-100% of rated speed. This change is intended to reduce the wear on the diesel generators and thereby enhance their overall reliability.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the emergency diesel generators overall reliability will be improved, and the change would not impair the ability of the diesel generator to respond to a fast start demand.

9.1.2 Abandonment of the Containment Penetration and Weld Channel Pressurization System

Description of Change:

RFC-DC-12-2895 abandoned portions of the Containment Penetration and Weld Channel Pressurization System.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that no additional equipment that may initiate a design basis accident was installed. Additionally, the limits on the containment leakage were unaffected, ensuring that the leakrates assumed in the accident analysis remained valid.

9.1.3 Modification of Secondary-side, In-line Chemistry Instrumentation

Description of Change:

RFC-DC-12-2915 modified the secondary chemistry instrumention. The condensate, feedwater, demineralized water, main steam and steam generator blowdown monitors were upgraded in both the turbine building laboratory and the nuclear sampling room.

Safety Evaluation Summary:

This change was reviewed, and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that this RFC upgrades existing equipment and does not change the operation of that equipment.

9.1.4 Separation of Auxiliary Feedwater Emergency Leakoff Lines

Description of Change:

RFC-DC-12-3043 modified the auxiliary feedwater pumps emergency leakoff lines. The modification consisted of providing separate flowpaths to a common three inch test line. The modification was made to preclude deadheading of an auxiliary feedwater pump due to adverse pump-to-pump interaction in response to NRC Bulletin 88-04, "potential safety-related pump loss." The modification also included the addition of orifices to limit the leakoff flow to 75 gallons per minute.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conslusion is based on the fact that the change makes modifications which improve the performance of equipment important to plant safety in that the change reduces the potential adverse impact of one train on the other.

9.1.5 Boron Injection Tank System Modification

Description of Change:

RFC-DC-12-3050 modified the Boron Injection Tank (BIT) and its associated piping after receiving approval from the NRC for a reduction in the tank's boric acid solution concentration. The line between the boron injection tank and the boric acid storage tanks was cut and capped, the boron injection tank flush lines were cut and capped, and the boron injection tank vent was modified.

Safety Evaluation Summary:

The changes to the BIT piping systems were reviewed and it was concluded that they did not represent an unreviewed safety question. A previously approved technical specification

amendment approved the change in the boric acid concentration. As a result, the boron injection tank no longer performs a safety function during a steam line break transient, the only accident affected by the boron reduction and deactivation of the BIT.

9.1.6 Removal of Obsolete Westinghouse Radiation Monitors

Description of Change:

RFC-DC-12-4078 removed radiation monitors, 1-R1, 2-R1 (control room area monitor) for Unit 1 and Unit 2, respectively, and 12-R3 (radio chemistry lab area monitor), and 1-R4, 2-R4 (charging pump room area monitor). These monitors were replaced with new Eberline monitors installed under RFC-2900 E.02.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the components have been replaced with upgraded equipment.

9.1.7 Use of Spare Containment Penetration as a Service Penetration

Description of Change:

RFC-DC-12-4122 removed the end caps on containment penetration number (CPN) 71 and installed flanges with blank covers in their place. This allows hoses and cables to enter the containment through CPN-71 rather than through the lower containment airlock during operational modes where containment integrity is not required, eliminating a personnel hazard.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the penetration will continue to meet its functional requirements, and this change does not adversely impact any safety related equipment.

9.2 PLANT MODIFICATIONS

9.2.1 Installation of Chemical Diffuser Headers in the Forebay

Description of Change:

Plant Modification (PM) 12-837 installed chemical diffuser headers in the forebay, upstream of the trash rack to evenly distribute biocidal chemicals in the forebay for zebra mussel control.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that implementing the PM will enhance the intake water supply by controlling or eliminating the growth of zebra mussels.

9.2.2 Modification of the PA System

Description of Change:

Plant Modification (PM) 12-1090 modified the plant PA system to help reduce background noise in the control rooms. At the time, paging could be addressed to either Unit 1, Unit 2, the office building, or the entire plant. As a result of the modification, paging can be addressed to either the control rooms, the office building, all areas of the plant except the control rooms and office buildings, or the entire plant.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion was based on the fact that the implementation of the PM would still allow the PA system to maintain its ability to perform its emergency plan functions.

9.2.3 New Project Engineering/Site Design Office Building Construction

Description of Change:

Plant Modification (PM) 12-1159 constructed a new office building for the Project Engineering and Site Design Groups. A new building was needed since the proposed site for the new fire protection water storage tanks is where the current offices are for the above mentioned groups. The selected location for this building is outside the protected area, south of the south security gate.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion was based on the fact that implementing the PM would not impact any safety-related equipment nor Seismic Class I structures that are important to safety and that the only impact to the UFSAR is in the plant layout drawings.

9.2.4 Radioactive Materials Building Tie-ins to Plant Systems

Description of Change:

Plant Modification (PM) 12-1190 proposed to make various tie-ins from the newly built radioactive materials building to the existing fire protection system, potable water system, sewer system and various plant alarm systems.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that implementing the PM would not prevent the above systems from performing their intended safety functions, if any, when called upon.

9.3 MINOR MODIFICATIONS

9.3.1 Relocation of Orifice for Turbine Driven Auxiliary Feedwater Pump

Description of Change:

Minor Modification (MM) 12-MM-242 relocated an orifice plate contained in the discharge piping of the turbine driven auxiliary feedwater pump. The orifice plate was moved further downstream and the associated instrument lines, isolation valves, and transmitters were also relocated.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that the change impacts only the location of the orifice plate and associated equipment. It does not impact the function of the auxiliary feedwater system nor does it degrade its structural integrity.

9.3.2 Modification of the Ice Machine Glycol Return and Supply Header

Description of Change:

Minor Modification (MM) 12-MM-267 modified the ice machine glycol supply and return header to accommodate supplemental cooling. The new installation required a tie-in utilizing a tee on a 6" section of pipe using a flange and installing a valve to isolate glycol flow.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety-question. This conclusion was based on the fact that the system is non-safety related and that the modification would not adversely impact systems important to safety.

9.4 TEMPORARY MODIFICATIONS

9.4.1 Close Control Room Ventilation Dampers 1,2-HV-ACRDA

Description of Change:

Temporary Modifications (TMs) 01-92-11 and 02-92-9 administratively maintain closed the control room normal intake damper (HV-ACRDA-1) for both units. This modification is in response to discussions held with the NRC regarding our control room habitability analyses that were submitted in letter AEP:NRC:0389-O on October 11, 1988. This letter submitted analyses of doses to control room operators following a LOCA. The analyses determined whole body doses that were within the 5 rem limit of General Design Criteria 19 of 10CFR50 Appendix A, and that were within the 50 rem thyroid and skin dose limits that were believed to be acceptable based on the International Commission on Radiological Protection, Publication Number 30. The NRC had informed us that they believe the correct limit for thyroid and skin dose is 30, rather than 50 rem. As a result, several measures were taken to align the control room ventilation system such that the 30 rem limit would not be exceeded during a postulated accident. These measures included maintaining closed the control room normal intake damper.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion was based on the fact that the change places the damper in the safe position for the types of events for which the control room ventilation system provides protection. The TM will stay in place until analyses can be completed to bring the control rooms limits for thyroid and skin doses to within the NRC required 30 rem.

9.4.2 Disconnect Inoperable Seismic Accelerometer

Description of Change:

Temporary Modification (TM) 01-92-40 involves disconnecting the power cables to seismic accelerometer No. 12-SMIC-ACL-2 (non-tech. spec) located on the top of the primary shield wall inside containment. This disconnection helps to restore the

natural frequency wave form of the remaining three accelerometers (required by technical specifications) until the faulty accelerometer can be replaced.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion is based on the fact that disconnecting the non-technical specification accelerometer will improve the performance of the other technical specification required accelerometers. During a design basis earthquake, sufficient information will be available from the accelerometers listed in the technical specification.