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FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana M 05000316

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AEP:NRC:1147B

Donald C. Cook Nuclear Plant Units 1 and 2
License Nos. DPR-58 and DPR-74
Docket Nos. 50-315 and 50-316
ANNUAL OPERATING REPORT

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

February 26, 1992

Dear Dr. Murley:

Paragraph 6.9.1.5 of the Donald C. Cook Nuclear Plant Technical Specifications requires that an annual report be submitted to address personnel exposure, steam generator in-service inspection results, challenges to power-operated relief and safety valves, and information regarding I-131 activity. In addition, 10 CFR 50.59(b)(2) requires submittal of an annual report describing changes, tests, and experiments made in the preceding year. Consistent with these requirements, attached are two copies of the Cook Nuclear Plant 1991 Annual Operating Report.

This document has been prepared following corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

E. E. Fitzpatrick

E. E. Fitzpatrick
Vice President

edg

Attachments
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ATTACHMENT TO AEP:NRC:1147B

DONALD C. COOK NUCLEAR PLANT

1991 ANNUAL OPERATING REPORT

DONALD C. COOK NUCLEAR PLANT
1991 ANNUAL OPERATING REPORT

February 28, 1992

COMPILED BY: W. R. Moran
W. R. Moran

REVIEWED BY: B. A. Svensson
B. A. Svensson
Executive Staff Assistant

APPROVED BY: E. E. Fitzpatrick
E. E. Fitzpatrick
Vice President-Nuclear Operations

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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. C. Loope	- Personnel Exposure Summary
C. A. Freer	- Steam Generator ISI Summary
B. A. Svensson	- Changes to Procedures
B. A. Svensson	- Challenges to Pressurizer PORVs and Safety Valves
B. A. Svensson	- Tests and Experiments
B. A. Svensson	- Reactor Coolant Specific Activity
D. H. Malin	- Results of Irradiated Fuel Inspections
R. G. Vasey	- Changes to Facility - RFCs, MMs, PMs
B. A. Svensson	- Changes to Facility - Temporary Modifications to Unit 1 & 2

2.0 PERSONNEL RADIATION EXPOSURE SUMMARY

Table 2 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 69.087 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

	# PERSONNEL >100 mR			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
Reactor Operations & Surveillance						
Maintenance Personnel	0007	0000	0004	001.308	000.000	001.285
Operations Personnel	0023	0001	0005	005.594	000.336	002.001
Health Physics Personnel	0026	0000	0026	008.514	000.000	008.108
Supervisory Personnel	0002	0000	0000	000.494	000.000	000.000
Engineering Personnel	0001	0000	0000	000.161	000.000	000.000
Routine Maintenance						
Maintenance Personnel	0014	0000	0031	002.796	000.000	008.659
Operations Personnel	0002	0000	0008	000.299	000.000	002.169
Health Physics Personnel	0003	0000	0022	000.763	000.000	005.704
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0001	0000	0000	000.572	000.000	000.000
In-Service Inspection						
Maintenance Personnel	0000	0000	0000	000.000	000.000	000.000
Operations Personnel	0000	0000	0000	000.000	000.000	000.000
Health Physics Personnel	0000	0000	0000	000.000	000.000	000.000
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0000	0000	0000	000.000	000.000	000.000
Special Maintenance						
Maintenance Personnel	0000	0000	0044	000.000	000.000	017.183
Operations Personnel	0000	0000	0004	000.000	000.000	000.808
Health Physics Personnel	0000	0000	0000	000.000	000.000	000.000
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0000	0000	0001	000.000	000.000	000.182
Waste Processing						
Maintenance Personnel	0001	0000	0001	000.154	000.000	000.512
Operations Personnel	0000	0000	0002	000.000	000.000	000.656
Health Physics Personnel	0000	0000	0004	000.000	000.000	000.830
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0000	0000	0000	000.000	000.000	000.000
Refueling						
Maintenance Personnel	0000	0000	0000	000.000	000.000	000.000
Operations Personnel	0000	0000	0000	000.000	000.000	000.000
Health Physics Personnel	0000	0000	0000	000.000	000.000	000.000
Supervisory Personnel	0000	0000	0000	000.000	000.000	000.000
Engineering Personnel	0000	0000	0000	000.000	000.000	000.000
TOTALS						
Maintenance Personnel	0017	0000	0071	004.258	000.000	027.639
Operations Personnel	0025	0001	0015	005.893	000.336	005.633
Health Physics Personnel	0026	0000	0047	009.277	000.000	014.642
Supervisory Personnel	0002	0000	0000	000.494	000.000	000.000
Engineering Personnel	0002	0000	0001	000.733	000.000	000.182
GRAND TOTALS	0072	0001	0134	020.655	000.336	048.096

3.0 STEAM GENERATOR IN-SERVICE INSPECTION

3.1 UNIT 1 INSPECTIONS

During 1991, there were no steam generator in-service inspections performed for Unit 1.

3.2 UNIT 2 INSPECTIONS

During 1991, there were no steam generator in-service inspections performed for Unit 2.

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 Procedure No. 12 MHP5021.001.007, Revision 6, C.S. 2, "Disassembly, Repair and Reassembly of Conval Clampseal Valves"

Description of Change:

This maintenance procedure change was implemented to permit the use of "Chesterton" valve packing on safety related valves in systems described in the UFSAR. The Chesterton valve packing differs from the valve packing configurations described in the UFSAR for certain safety related valves.

Safety Evaluation Summary:

A technical evaluation determined that the packing system described in the UFSAR has been rendered obsolete by advances in technology. The Chesterton system of packing currently being installed in the majority of the safety related valves is a direct development of an EPRI study and is technically superior to that described in the UFSAR.

The safety evaluation determined that the changes do not involve an unreviewed safety question as defined in 10 CFR50.59. Changes to the UFSAR will be incorporated in the next annual update.

4.1.2 Procedure No. 2-THP4030.STP.510, Revision 2, C.S. 16, - "Reactor Trip SSPS Logic and Reactor Trip Breaker Train "A" Surveillance Test (Monthly)

Description of Change:

The change sheet to the above procedure was written to allow the use of backup manual testing for the solid state protection system (SSPS). The change was necessary because the semi-automatic tester, which is described in the UFSAR, appears to be malfunctioning and providing ambiguous test results for some logic circuits. The change to allow manual testing is a change to a procedure described in the UFSAR. The SSPS was designed for either automatic or manual testing.

Safety Evaluation Summary:

The changes do not affect the function of any safety related equipment. The safety evaluation concluded that the changes to the procedure do not constitute an unreviewed safety question as defined in 10 CFR50.59.

4.2 OPERATING PROCEDURES

4.2.1 Procedure No. 1-OHP4021.028.017, Revision 1 Procedure No. 2-OHP4021.028.017, Revision 0 - "High Velocity Flush of Containment Lower Compartment Ventilation Units and RCP Motor Air Coolers"

Description of Change:

The above procedures were implemented for the purpose of performing high velocity flushes of the containment lower compartment ventilation units and the RCP motor air coolers. Chapter 5.5 of the UFSAR states that the maximum non-essential service water flow to the lower compartment ventilation units is 440 gpm. During the flushing procedure, the flow will be greater than 440 gpm.

Safety Evaluation Summary:

Section 9.8.3.2 of the UFSAR describes the containment lower compartment ventilation units and the RCP motor air coolers as components having no safety-related functions. The safety evaluation concluded that implementation of these procedures will not involve an unreviewed safety question, as defined in 10 CFR50.59.

4.2.2 Procedure No. 12 OHP4021.057.008, Revision 0 - "Operation of the Circulating Water System to Increase Forebay Temperatures"

Description of Change:

The above procedure was implemented for the purpose of operating the circulating water system in a manner which increases the forebay temperature above the lake temperature. The procedure will be used for the purpose of activating zebra mussels so that a mulluscicide treatment can be performed.

Safety Evaluation Summary:

The review focused on the impact of the change on the essential service water system, since it is the only safety-related system directly impacted by the procedure. The procedure contains a precaution that the temperature of the condenser inlet water cannot exceed 87°F. The unreviewed safety question determination concluded that the procedure does not constitute an unreviewed safety question as defined in 10 CFR50.59.

4.2.3 Procedure No. 1-OHP4024.102, Revision 4, C.S. 6, "Annunciator #102 Response; Miscellaneous Areas Fire System"

Description of Change:

Annunciator #102 Response procedure was changed to reflect the removal of the Halon fire suppression system from the office

building fourth floor Q/C vault. The change in the Halon fire suppression system was implemented under Plant Modification No. 1-PM-744. The area is no longer being used as a Q/C vault.

Safety Evaluation Summary:

The portion of the Halon system being removed does not affect nuclear safety. The safety evaluation concluded that removal of the Q/C vault Halon system does not involve an unreviewed safety question as defined in 10 CFR50.59.

4.2.4 <u>Procedure No. 2-OHP SP.092, Revision 0 - "Raising Reactor Power with High Feedwater Delta-P Signal"</u>
<u>Procedure No. 2-IHP6030.IMP.422, Revision 0, C.S. 2 - "Main Feedpump Speed Control Calibration/Alignment"</u>

Description of Change:

A problem had developed with steam generator #21 feedwater regulating valve, 2-FRV-210, sticking when the valve was open greater than 60%. A temporary solution was to operate with the regulating valves to all four steam generators in a less open position. This could be accomplished by increasing the differential pressure across the valves by operating the feedwater pumps at a higher speed. The total feedwater flow to the steam generators would remain the same.

Special Procedure 2-OHP SP.092, Revision 0 was implemented to allow raising reactor power from approximately 90% to 100% while operating with the elevated feedwater differential pressure and collecting data for establishing the revised differential pressure program. The new differential pressure program would be determined based on the feedwater regulating valves operating approximately 50% open at 100% power.

Procedure 2-IHP6030.IMP.422 was changed to implement the recalibration of the differential pressure program controller to the new values.

Safety Evaluation Summary:

Operation of the feedwater regulating valves as described above may result in a maximum flow exceeding that used in the safety analysis for a failed open feedwater regulating valve. The UFSAR contains an analysis of a feedwater regulating valve malfunction using a flow of 150% of nominal. Thus operation in this mode will result in a change to the facility as described in the UFSAR.

The safety evaluation was performed assuming a maximum flow of 200% of nominal, a value judged to be bounding. A feedwater malfunction is evaluated against departure from nucleate boiling (DNB) criteria (a minimum ratio of 1.38 to 1.607, depending on the location). The evaluation concluded that the

DNB criteria are met with a flow of 200% of nominal. The unreviewed safety question determination concluded that implementation of this procedure does not constitute an unreviewed safety question as defined in 10 CFR50.59.

4.3 CHEMISTRY PROCEDURES

4.3.1 Procedure No. 12-THP6020.LAB.041, Revision 16 - "Data Sheet Instructions"

Description of Change:

One of the changes implemented by Revision 16 involves an increase in the upper limit of Lithium-7 (Li-7) concentration in the reactor coolant system (RCS) recommended by Westinghouse which will require a change to the UFSAR. Specifically, the UFSAR gives a range of RCS Li-7 concentration of 0.22 to 2.2 ppm, whereas the new proposed range received from Westinghouse is 0.20 to 3.8 ppm.

Safety Evaluation Summary:

An increase in the Li-7 concentration in the RCS can theoretically increase the amount of tritium produced in the primary system. The Li-7 is introduced to the RCS in the form of lithium hydroxide, which is used in controlling primary system pH.

An evaluation of the impact on RCS tritium inventory that could result from a higher Li-7 concentration in the RCS was performed. The evaluation concluded that, from an operational standpoint, the proposed increase in the allowable range of RCS Li-7 concentration would still result in total tritium activity in the RCS well below the Li-7 concentration limit of 2.51Ci/cc given in the UFSAR. The safety evaluation concluded that increasing the allowable range of RCS Li-7 concentration will not increase the dose consequences of an accident and that the change does not involve an unreviewed safety question as defined in 10 CFR50.59.

4.4 ENVIRONMENTAL MONITORING PROCEDURES
PROCEDURE NO. 12-THP6010.ENV.065, REVISION 1, - "COLLECTION OF DRINKING WATER SAMPLES FROM PUBLIC INTAKES"

Description of Change:

The above procedure was changed to reflect changes in the Technical Specification requirements for obtaining drinking water samples as part of the routine radiological environmental monitoring program. The revision deleted two sample locations (Benton Harbor and New Buffalo) which were still identified in the then current Emergency Plan. The Emergency Plan and implementing procedures have since been changed to reflect the current NRC approved sample locations.

Safety Evaluation Summary:

The procedure changes apply only to the collection of drinking water samples as part of the radiological environmental monitoring program for normal plant operations, and were implemented after receiving NRC approval via technical specification Amendment 94 for Unit 1 and Amendment 80 for Unit 2. The safety evaluation concluded that implementation of the changes does not represent an unreviewed safety question as defined in 10 CFR50.59.

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.

During 1991, there were no procedures classified as a "Test and Experiment" implemented.

6.0 CHALLENGES TO PRESSURIZER POWER OPERATED RELIEF VALVES AND SAFETY VALVES

During 1991, 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 during 1991.

7.0 REACTOR COOLANT SPECIFIC ACTIVITY

During 1991, 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 1991, there were no examinations performed on the irradiated fuel due to not having any scheduled refueling outages.

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 Removal of Obsolete Westinghouse Monitor

Description of Change:

RFC-DC-12-4078 removed Westinghouse and NMC radiation monitors which had been replaced by Eberline units, but left in place. The monitors were R.25/26 (unit vent stacks air particulate), R.2 (upper containment area), R.33 (turbine gland steam condenser vent) and R.31/32 (unit vent stack effluent iodines and noble gases). The change consisted of removing cabinets and disconnecting electrical leads.

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 simply removes equipment that has been replaced by another brand. The function of the system has not been changed.

9.2 Removal of Axial Power Distribution Monitoring System

Description of Change:

RFC DC-12-2937 removed the axial power distribution monitoring system components from their control room racks and removed the system annunciators.

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 subject system was only a monitoring system that was not used for any safety related function but which had once been referred to in the Technical Specifications. The NRC had approved Technical Specification changes which had deleted the reference to this system.

9.3 Replacement of Liquid Effluent Monitor R-18

Description of Change:

RFC DC-12-2853 replaced an existing monitor with a more sensitive monitor.

The R-18 monitor is used to monitor the release of liquid effluents from the waste disposal system. Should the discharge exceed a preset limit, R-18 provides a signal which closes the discharge valve.

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 function of R-18 remains the same and the increased sensitivity of the replacement monitor enhances the system's capabilities.

9.4 Removal of Accumulator Backup Nitrogen Supply

Description of Change:

PM 12-491 removed a four-bottle backup nitrogen supply from the accumulator nitrogen space. The backup supply was normally isolated from the accumulator, with nitrogen still supplied to the accumulators (an intermittent process) by the main nitrogen system, a six tank bank.

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 nitrogen system performs no active function during an accident, and a nitrogen supply to the accumulator continues to exist.

9.5 Replacement of Automatic Gas Analyzer

Description of Change:

RFC DC-12-2591 replaced the automatic gas analyzer sampling panel, connected the new panel to the nuclear sampling room ventilation system and modified a water drain line. The automatic gas analyzer is used to measure oxygen and hydrogen concentrations in the gas decay tanks. Its replacement was necessary because major components were obsolete and could not be repaired.

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 involved the replacement of existing equipment with comparable equipment performing the same function.

9.6 Removal of Failed Fuel Detector

Description of Change:

RFC DC-12-2911 involved removal of the failed fuel detector (FFD) because of difficulties in maintaining the calibration of the system. The changes consisted of cutting and capping of lines, removal and disposal of the failed fuel detector and associated instrumentation, disconnecting the power supply to the FFD and removal of computer inputs from the FFD.

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 FFD is a monitoring device and its function can be accomplished by the analysis of periodically obtained grab samples. The use of grab samples had been approved by the NRC in response to our Regulatory Guide 1.97 deviation request submitted via letter AEP:NRC:0773AB.

9.7 Removal of Local Halon Fire Suppression System

Description of Change:

PM 1-744 removed the Halon tanks from a fire suppression system in a former Quality Control (QC) records vault in the office building. Because of several concerns with the use of the fluorocarbon Halon, its use is being minimized at the Cook Nuclear Plant. The room associated with the bottles being removed was no longer used to store QC records, and the Halon suppression system was no longer required in that area.

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 no safety-related equipment was impacted by the removal of the Halon system from the office building area.

9.8 Replacement of U-1 Moisture Separator Reheater Tube Bundles

Description of Change:

PM 1-803 replaced the Unit 1 reheater tube bundles with an improved design, installed perforated plates and improved chevrons in the moisture separator, and modified the moisture separator drain lines.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it does not represent an unreviewed safety question. This conclusion was based on the fact that the components which were modified are non safety related and the changes did not impact any safety related components or systems.

9.9 Install Artificial Leak By on SI Pump Discharge Check Valve

Description of Change:

MM 2-166 installed a bypass line around the south safety injection (SI) pump discharge check valve. This modification prevents pressure buildup in the SI pump headers (caused by leaking check valves) by allowing a small quantity of fluid to flow through the SI pump miniflow line to the refueling water storage tank.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it does not represent an unreviewed safety question. This conclusion was based on the fact that the modification does not impair the capability of the SI pumps to provide adequate cooling water flow in the event of an accident.