ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8903100329 DOC.DATE: 88/12/31 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

AUTH.NAME	AUTHOR AFFILIATION
STEVENSON,B.A.	Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
GIBSON,L.S	Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
SMITH,W.G.	Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: "Annual Operating Rept,1988." W/890228 ltr.

DISTRIBUTION CODE: IE47D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 20
TITLE: 50.59 Annual Report of Changes,Tests or Experiments Made W/out Approv

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-1 LA	1	0	PD3-1 PD	5	5
	STANG,J	1	0			
INTERNAL:	AEOD/DOA	1	1	AEOD/DSP/TPAB	1	1
	NRR/DLPQ/HFB 10	1	1	NRR/DOEA/EAB 11	1	1
	~~NRR/DREP/RPB~~ 10	2	2	NUDOCS-ABSTRACT	1	1
	~~REG FILE~~ 02	1	1	RGN3 FILE 01	1	1
EXTERNAL:	LPDR	1	1	NRC PDR	1	1
	NSIC	1	1			

Add: NMSS/Imsb 1 1
RES Brook, B 1 1

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~19~~ 21 ENCL ~~17~~ 19

DONALD C. COOK NUCLEAR PLANT

ANNUAL OPERATING REPORT

1988

COMPILED BY: B.A. Svensson
Licensing Activities Coordinator

REVIEWED BY: L.S. Gibson
Asst. Plant Mgr.-Technical Support

APPROVED BY: W.G. Smith, Jr.
Plant Manager

Add: NMSS/IMSB
RES Brook, B

IE47
1/1

TABLE OF CONTENTS

TITLE	PAGE NUMBER
Introduction	1
Personnel Exposure Summary	2
Steam Generator Inservice Inspection Reports . . .	5
Changes to Facility	6
Changes to Procedures	15
Challenges to Pressurizer PORVS and Safety Valves	16
Irradiated Fuel Inspection	17

INTRODUCTION

The Donald C. Cook Nuclear Plant, owned by Indiana Michigan Power Company is located five miles north of Bridgman, Michigan and consists of two nuclear power units. Each unit employs a pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation.

The Unit 1 reactor is currently designed for a power output of 3250 MWt and the Unit 2 reactor is designed for a power output of 3411 MWt, which are their licensed ratings. The approximate gross and net electrical outputs of Unit 1 are 1056 MWe and 1020 MWe and of Unit 2 are 1100 MWe and 1060 MWe, respectively. The main condenser cooling method is open cycle using Lake Michigan water as the cooling source. The Cook Nuclear Plant was the first domestic nuclear facility to employe the ice condenser reactor containment system. The American Electric Power Service Corporation was the architect-engineer and constructor.

This Report was compiled by B.A. Svensson with the following individuals contributing information as follows:

D.C. Loope - Personnel Exposure Summary

C.A. Freer - Steam Generator ISI Summary

S.D. DeLong - Changes to Facility

J.B. Droste - Changes to Procedures

B.K. Wonn - Challenges to Pressurizer PORVs and Safety Valves

R.W. Hennen - Results of Irradiated Fuel Inspections

PERSONNEL EXPOSURE SUMMARY

ANNUAL OPERATING REPORT - RG 1.16 for 1988

	# PERSONNEL > 100mR			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
REACTOR OPERATIONS & SURVEILLANCE						
Maintenance Personnel	0007	0000	0010	0000.971	0000.000	0003.062
Operations Personnel	0056	0001	0033	0015.451	0000.155	0013.699
Health Physics Personnel	0018	0000	0061	0007.353	0000.000	0022.952
Supervisory Personnel	0000	0000	0001	0000.000	0000.000	0000.130
Engineering Personnel	0010	0000	0004	0001.822	0000.000	0000.674
ROUTINE MAINTENANCE						
Maintenance Personnel	0101	0002	0241	0044.339	0000.267	0155.889
Operations Personnel	0019	0001	0022	0005.910	0000.338	0006.246
Health Physics Personnel	0006	0000	0013	0001.205	0000.000	0005.106
Supervisory Personnel	0002	0000	0001	0000.285	0000.000	0000.244
Engineering Personnel	0003	0003	0000	0000.357	0000.536	0000.000
IN-SERVICE INSPECTION						
Maintenance Personnel	0005	0000	0018	0000.954	0000.000	0003.448
Operations Personnel	0003	0001	0014	0000.630	0000.650	0008.586
Health Physics Personnel	0003	0000	0003	0000.494	0000.000	0000.929
Supervisory Personnel	0001	0000	0000	0000.170	0000.000	0000.000
Engineering Personnel	0000	0000	0001	0000.000	0000.000	0000.170
SPECIAL MAINTENANCE						
Maintenance Personnel	0011	0031	0680	0001.722	0028.997	0541.633
Operations Personnel	0020	0010	0011	0004.961	0004.987	0010.332
Health Physics Personnel	0000	0000	0079	0000.000	0000.000	0103.184
Supervisory Personnel	0010	0010	0012	0001.102	0000.432	0004.257
Engineering Personnel	0010	0010	0020	0000.145	0003.130	0022.918
WASTE PROCESSING						
Maintenance Personnel	0001	0000	0054	0000.620	0000.000	0019.097
Operations Personnel	0000	0000	0004	0000.000	0000.000	0002.429
Health Physics Personnel	0000	0000	0017	0000.000	0000.000	0004.597
Supervisory Personnel	0000	0000	0000	0000.000	0000.000	0000.000
Engineering Personnel	0001	0000	0001	0000.170	0000.000	0000.175
REFUELING						
Maintenance Personnel	0004	0000	0018	0000.528	0000.000	0008.664
Operations Personnel	0010	0001	0066	0004.750	0000.106	0033.389
Health Physics Personnel	0002	0000	0014	0000.205	0000.000	0005.743
Supervisory Personnel	0000	0000	0000	0000.000	0000.000	0000.000
Engineering Personnel	0001	0001	0000	0000.121	0000.151	0000.000
TOTALS						
Maintenance Personnel	0102	0002	0324	0047.532	0000.267	0192.357
Operations Personnel	0073	0002	0106	0026.741	0001.249	0064.458
Health Physics Personnel	0018	0000	0079	0009.257	0000.000	0039.327
Supervisory Personnel	0003	0000	0001	0000.455	0000.000	0000.484
Engineering Personnel	0011	0004	0005	0002.470	0000.687	0001.019
GRAND TOTALS	0511	0079	1913	0172.910	0004.406	0595.290

1.16 REPORT - WORK FUNCTION CATEGORIES

Reactor Operations and Surveillance

Those activities involved with operating the plant or monitoring it's operation, including chemistry, performance testing, surveillance testing, etc. The plant may be at any power level, including zero, and still have work falling into this area. Many STP's run during shutdown or refueling may still fall into this category.

Routine Maintenance

All equipment or system maintenance, whether preventative or restorative, which does not involve significant modifications to equipment or systems. Included is I&C repair work, as well as work to adjust operable equipment to improve performance (adjusting fan blade pitch, for example).

Inservice Inspection

Inspections of equipment and systems to monitor changes that would be detrimental to function or integrity. Also included is all work required to permit such inspections, such as building required scaffolding, removing or replacing supports of insulation, or disassembly of valves, pumps, etc. Not included are inspection to assess or monitor normal wear, etc. For example, dissembly of a charging pump to inspect bearing wear would not be Inservice Inspection, but dissembly to inspect for rotor cracking or casing damage would be. Inspection of a weld on a newly added line is Special Maintenance, or inspection of a weld repair at a leaking fitting is Routine Maintenance.

Special Maintenance

All work on equipment or systems performed to make significant modifications. Installation of new systems or equipment, replacement or addition of supports or hangers, addition of new lines or instruments, removal of existing equipment, replacement of existing equipment with significantly different equipment are all Special Maintenance. For example, replacement of a properly functioning, original equipment pressure transmitter with a different model with improved characteristics or certification would be Special Maintenance, but replacement or a malfunctioning pressure transmitter with a newer or improved model would probably be Routine Maintenance.

Waste Processing

All work associated with decontamination of equipment, areas, systems, etc. (if not an integral part of another job, such as pump repair), collection and processing of waste, whether solid, liquid, or gas. Operations in support of waste handling are also included. For example, draining a filter to permit changing it, or venting it after changing are part of Waste Processing, but valving a clean filter into the system is Reactor Operations. Repair of the Baler or drumming room crane is Routine Maintenance.

Refueling

All work is directly concerned with refueling the reactor, including all support operations, is classified as Refueling. Testing the polar crane or installing the cavity filter rig is part of Refueling, as is cavity decon before or after flood-up. Changing the cavity filter, however, is Waste Processing and fixing the manipulator crane is Routine Maintenance.

STEAM OPERATOR TUBE INSERVICE INSPECTION REPORTS

1988 SUMMARY REPORTS

UNIT NO. 1

There were no inservice inspections of Unit No. 1's steam generators for the year 1988.

UNIT NO. 2

Unit No. 2 was removed from service on April 23, 1988, for the complete replacement of the four steam generator lower assemblies. A complete "preservice inspection" of the new lower assemblies was performed following the field hydrostatic test and prior to initial operation using equipment and techniques expected to be used during subsequent inservice inspection, pursuant to the requirements of Technical Specification 4.4.5.4.a.9.

There were no reportable indications identified during the preservice inspection.

REQUEST FOR CHANGE
CHANGES TO FACILITY

Brief descriptions and summary safety evaluations for design changes (RFCs) made to the facility as described in the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR) are presented in this section. These changes were completed pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59 (a).

RFC-12-2859

BRIEF DESCRIPTION

RFC-DC-12-2859 provided for the replacement of the South Boric Acid Evaporator Steam Coil Tube Bundle. Eddy current testing of the original tubes indicated severe pitting of the tubes. This pitting was most likely caused by Chlorides present in the evaporator bottoms. Although originally installed as a Boric Acid Evaporator, this evaporator is currently being used to process radwaste which has much higher chloride concentrations that can be expected in boric acid service. Based on recommendations from the Vendor, the tube bundle was replaced with a tube bundle manufactured from Incoloy 825 material. It is expected that the Incoloy 825 will perform better than the originally supplied 304SS bundle because of its superior resistance to chloride pitting.

SAFETY EVALUATION

This RFC has been classified as Safety-Interface since the South Boric Acid Evaporator is a Seismic Class II component.

Nuclear Safety and Licensing has reviewed the change as per the review criteria in NS&L procedure No. 7. As a result of the Safety Review, there were no open items for this RFC. The Incoloy material was determined to have higher allowable stress limits than the stainless steel material.

The purpose of this review was for procurement, design and installation. It was concluded, by the review, that this RFC did not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor did it create a substantial hazard to the health and safety of the public.

RFC-12-2908 (Addendum #3)

BRIEF DESCRIPTION

RFC-DC-12-2908 (Addendum #3) provided for the installation of a new 150 ton/ 20 ton Auxiliary Building Crane and bridge system. This crane was installed on the existing crane rails and was used in tandem with the existing crane that was modified under RFC-DC-12-2962 (also included in this submittal). Together, these cranes were used to lift and move the steam generators through the auxiliary building during the Steam Generator Replacement Project.

Section 9.7 of the FSAR has been revised to incorporate the changes made to the auxiliary building crane. Additional information concerning the cranes may be found in that section of the FSAR.

SAFETY EVALUATION

This RFC has been classified as safety related because it involves modifications to the auxiliary building crane which is Seismic Class I equipment.

In addition to the safety evaluation provided for RFC-DC-12-2962, it was necessary to evaluate the consequences of a load drop during the installation of the crane components. Items covered were: (1) the individual crane components while they were being lifted by a boom crane, and (2) inadvertently dropping the boom crane components onto the auxiliary building structural elements or inadvertently hitting the auxiliary building structural elements. These evaluations were performed by the AEPSC Structural Design section.

Nuclear Safety and Licensing has reviewed these evaluations and found them to be acceptable with the following comments on the boom evaluation:

a) The boom crane has a capacity of 600K and the maximum weight lifted was that of the trolley at 144K. The safety factor available was 4.17. This was reasonably high for handling an occasional load.

b) It was noted that, in order to avoid any type of inadvertent human error, an additional operating engineer was available who also worked as an "oiler." It was recommended that a dedicated operating engineer be posted without any other assignments that would demand his attention.

c) After discussion with the AEPSC Material Handling Division, it was Nuclear Safety and Licensing's understanding that the maximum load lifted during the crane installation activity was 72 tons and the boom crane was certified to carry a test load of 110% of the maximum load as per ANSI B30.5. After the boom crane was installed at the site, an installation certificate was issued to document that the boom crane was installed per the manufacturer's guidelines.

d) The procedures for the crane installation were reviewed and approved by the appropriate engineering disciplines in AEPSC.

Based on the evaluation noted above, it is concluded that the installation of the cranes does not constitute an unreviewed safety question as per 10 CFR 50.59 Section (a)(2) and will not adversely affect the health and safety of the public.

RFC-12-2962

BRIEF DESCRIPTION

RFC-DC-12-2962 provided for modifications to the auxiliary building crane in order to conform to the single-failure-proof requirements of NUREG-0554. Specifically, the modification involved (a) replacing the original trolley (150T/20T) with a new trolley (150T) designed and built to single-failure-proof (SFP) requirements, (b) adding a second holding brake and an inching mechanism to the bridge drive, and (c) upgrading the crane runway girder in the auxiliary building to resist the higher wheel loads.

These modifications were performed for two reasons: (1) The crane must meet the requirements of NUREG-0612 and 0554 in regard to handling of heavy loads in the auxiliary building. (2) The single-failure-proof features of the modified crane were necessary for the movement of steam generators in and out of the auxiliary building during the S/G replacement project.

Section 9.7 of the FSAR has been revised incorporating the changes made to the auxiliary building overhead crane. Reference the revised text in Section 9.7 should any additional information be desired.

SAFETY EVALUATION

The auxiliary building crane is a Seismic Class I component which performs various Safety-Related activities such as opening/closing of the containment equipment hatches and moving new and spent fuel assemblies and lifting and transporting steam generators during their replacement activities. Therefore, this RFC was classified as Safety-Related.

The safety memo specifically addressed the following items related to the crane modifications:

1. The modifications to the crane as noted in the original RFC have been completed. Specifically, (a) the new trolley has been designed and fabricated to meet the single-failure-proof requirements of NUREG-0554, (b) a second holding brake and an inching mechanism has been added to the bridge drive and (c) the crane runway girder has been modified to take the higher wheel loads.

2. NS&L has reviewed the stress/seismic analysis of the modified crane performed by Whiting and found it to be acceptable.

3. AEPSC Cognizant Engineers have visited the Whiting offices to review the vendor documents as per the requirements of specification DCC-MH-105-QCN and accepted the new trolley.

4. The auxiliary building crane is a seismic Class 1 component and all modifications to the crane have been procured and installed to meet the seismic Class 1 requirements.

Based on the evaluation described above, it is concluded that the design modification performed to the auxiliary building crane does not constitute an unreviewed safety questions as per 10 CFR 50.59, Section (a) (2) and that it will not adversely affect the health and safety of the public.

The NRC has reviewed the modifications to the auxiliary building crane in amendment 100 to facility operating license No. DPR-74. Their concurrence reads as follows:

> "Based on the licensee's demonstration of compliance to the guidelines of NUREG-0554 and the adequacy of the crane's structural component in meeting their allowable stress values, the staff finds that the proposed new crane installation is acceptable."

<u>RFC-DC-12-4042</u>

<u>BRIEF DESCRIPTION</u>

RFC-4042 provided for the installation of an additional level indicator and low-level alarm for the Volume Control Tank (VCT) level control system. These modifications were installed to eliminate an undesirable situation should one of the two (2) originally installed VCT level channels fail.

A failure of the capillary reference leg on the VCT Level Controller (QLC-452) will cause the instrument to fail high. This high level signal causes the VCT Divert Valve (QRV-303) to open in an effort to reduce and maintain the VCT level within a normal operating band. As long as the Redundant VCT Level Controller (QLC-451) functioned normally, no alarm would have sounded.

This scenario had the potential of allowing the VCT to be pumped down until the charging pumps lost suction. The additional low level alarm added under this RFC is fed from QLC-451. This alleviates any concerns regarding the scenario described above.

<u>SAFETY EVALUATION</u>

This design change was an NRC commitment to provide additional VCT level instrumentation to aid plant operations. Since this modification will enhance the safety function of the VCT, this RFC has been classified as "Safety Interface".

This RFC has been reviewed in accordance with NS&L procedure number 7 "Safety Review of Design Changes". Subsequent conversations with the I&C engineer permitted a conclusion that there were no open items with regard to RFC-DC-12-4042 and the modifications described above did not constitute an unreviewed safety question as defined in 10 CFR 50.59.

MINOR MODIFICATIONS
CHANGES TO FACILITY

Brief descriptions and summary safety evaluations for design changes (Minor Modifications) made to the facility as described in the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR) are presented in this section. These changes were completed pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59 (a).

<u>12-MM-010</u> REMOVE WASTE EVAPORATOR FILTER ELEMENTS

<u>BRIEF DESCRIPTION</u>

This Minor Modification removed the Waste Evaporator Filter elements from the Waste Evaporator Feed Filters. Most liquid radwaste at the plant is currently treated with the Duratek demineralization system rather than the radwaste evaporators. Although Duratek has its own filters prior to the demineralizers, wastes are currently also being passed through the waste evaporator filters. It has been determined that this extra filtration is not needed to obtain adequate clean-up with the Duratek system. The waste evaporator filters are difficult to change and result in worker exposures that are not consistent with the principles of ALARA.

<u>SAFETY EVALUATION</u>

This change is classified as Safety Interface because it involves a system that handles radioactive wastes.

A technical evaluation was performed by the Chemical Engineering section. Specifics of the safety review for the modifications are provided below.

1. The filter units are designated seismic Class II. Operation without the removable filter elements in place in the filter housing will not degrade the seismic rating of the piping.

2. All liquid effluents are routed to the monitor tanks, where they are sampled prior to discharge to the lake. The sampling ensures that the effluents will not violate 10 CFR 20 or Technical Specification limits. Thus, even if cleanup capability is reduced by removal of the filter elements, it would not result in releases exceeding those limits.

3. The waste evaporator filters are discussed in Chapter 11.1 of the FSAR.

The safety evaluation concluded that removal of the waste evaporator filter elements does not constitute and unreviewed safety question as described in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2) and that it does not significantly impact public health and safety.

TEMPORARY MODIFICATIONS
CHANGES TO FACILITY

Brief descriptions and summary safety evaluations for Temporary Modifications made to the facility as described in the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR) are presented in this section. These changes were completed pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59 (a).

Temporary Mod. #19 (Unit 2)

BRIEF DESCRIPTION

This Temporary Modification involves the lifting of Cables 3167C-2, 3168-1, 3153C-2, 3138-2 and 3139-2 which is the power feed for pressurizer heaters #3, #4, #53, #55 and #56, respectively. These pressurizer heaters have a defective heater element. By disconnecting these cables, pressurizer heaters #3, #4, #53, #55 and #56 will not be able to perform their intended function.

SAFETY EVALUATION

This Temporary Modification has been classified as safety related because it affects the Reactor Coolant System.

The Plant Nuclear Safety Review Committee (PNSRC) has reviewed this Temporary Modification per the review criteria of PMI-1040, Rev. 3

It was concluded by the review, that this Temporary Modification does not constitute an unreviewed safety questions as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the public.

Temporary Mod. #35 (Unit 1)

BRIEF EVALUATION

This Temporary Modification involves the lifting of Cable 3151-1 which is the power feed for pressurizer heater #48. This pressurizer heater has a defective heater element. By disconnecting this cable, pressurizer heater #48 will not be able to perform its intended function.

SAFETY EVALUATION

This Temporary Modification has been classified as safety related because it affects the Reactor Coolant System.

The Plant Nuclear Safety Review Committee (PNSRC) has reviewed this Temporary Modification per the review criteria of PMI-1040, Rev. 3.

It was concluded by the review, that this Temporary Modification does not constitute an unreviewed safety questions as defined in 10 CFR 50.59 nor does it create a substantial hazard to the health and safety of the public.

Temporary Mod. #36 (Unit 1)

BRIEF DESCRIPTION

This Temporary Modification involves the lifting of Cable 3139-1 which is the power feed for pressurizer heater #56. This pressurizer heater has a defective heater element. By disconnecting this cable, pressurizer heater #56 will not be able to perform its intended function.

SAFETY EVALUATION

This Temporary Modification has been classified as safety related because it affects the Reactor Coolant System.

The Plant Nuclear Safety Review Committee (PNSRC) has reviewed this Temporary Modification per the review criteria of PMI-1040, Rev. 3.

It was concluded by the review, that this Temporary Modification does not constitute an unreviewed safety question as defined in 10 CFR 50.59 nor does it create a substantial hazard to the health and safety of the public.

Temporary Mod. #43 (Unit 1 & 2)

BRIEF DESCRIPTION

Installation of the Duratek Demineralization System using a mechanical jumper to route waste hold-up tank water to the 587' drumming room for processing. The effluent will be routed to the waste evaporator condensate tanks. All hose connections will have a working pressure at 300 PSI.

The system adds additional waste processing capability with a maximum feed flow of 55 GPM and will be used as an alternative to South Radwaste Evaporator operation.

SAFETY EVALUATION

This Temporary Modification has been classified as Safety Interface because this system handles radioactive solids, liquids and gases. The system itself is entirely Seismic Class III.

The Nuclear Safety and Licensing Section has reviewed this proposed change as per the review criteria in NS&L Procedure No. 7. As a result of the Safety review, it was concluded that this Temporary Modification does not constitute an unreviewed safety question as defined in 10 CFR 50.59 nor does it create a substantial hazard to the health and safety of the public.

Temporary Mod. #149 (Unit 1 & 2)

BRIEF DESCRIPTION

This Temporary Modification involves the addition of a 15% Sodium Hypochlorite solution into the Circulating Water supply in lieu of the Chlorination System to control algae and slime and to regain cooling efficiency in the Circulating Water Condensers.

SAFETY EVALUATION

The subject modification has been classified as Safety Interface since it involves adding chemicals that may interact with the Essential Service Water System, a Class I system.

It was concluded that the addition of Sodium Hypochlorite would not adversely affect the safety systems of the Plant. Further, it was pointed out that one of the reasons the system is described in the FSAR is that the NRC is the lead Federal Agency and addition of Sodium Hypochorite to the lake is an environmental matter normally handled by the Environmental Protection Agency.

This Modification does not constitute an unreviewed safety question as defined in 10 CFR 50.59 and will not adversely affect the health and safety of the public.

CHANGES TO PROCEDURES

A brief description of a procedure change implemented under the provisions of 10 CFR 50.59 and the associated safety evaluation is provided below:

Unit 2 Main Steam Safety Valve Set Point Verification Procedure

The change in Unit 2 main steam safety valve set point verification consisted of a new procedure implemented through Special Procedure 12 MHP SP.126, Revision 1, which allows setpoint testing of main steam safety valves in Modes 1, Power Operation, through Mode 3, Hot Standby. One safety valve is tested at a time. During testing the valve is considered inoperable per T/S 3.7.1.1. Thus per the Action Statement, the Power Range Neutron Flux High Setpoint is reduced per Table 3.7-1. If a problem should occur, the most plausible problem with the test would a safety valve sticking open. Since only one valve at a time is tested, any transient resulting from a stuck-open valve is bounded by the Unit 2 steam line break analysis found in Section 14.1.5 of the updated FSAR. This conclusion is valid in Modes 1, 2, and 3. Accident analysis assumptions which rely on the safety valve to open to relieve pressure, are maintained by causing a stuck-open safety valve during power operation, there is a possibility of a reactor trip. Although this potential exists, its possibility is minimized by provided a hydraulic closing device. Even if a trip occurs the consequences are bounded buy the existing accident analyses.

A change to Section 10.2.4 of the Updated FSAR has been made which states that "steam generator safety valve setpoints are checked periodically prior to or during scheduled outages".

The safety evaluation concluded that testing of the Unit 2 main steam safety valves in Modes 1, 2, or 3 does not constitute an unreviewed safety question as defined in 10 CFR 50.59.

CHALLENGES TO PRESSURIZER PORV'S AND SAFETY VALVES

There were no challenges to the Pressurizer PORV's or Safety Valves for either Unit 1 or 2 of the Donald C. Cook Nuclear Plant during 1988.

Annual Operating Report - Irradiated Fuel Examinations

During 1988, two separate examinations were performed on the irradiated fuel discharged from Unit 2 Cycle 6. These examinations were conducted in parallel with, or shortly after, the core was unloaded, and the intent was to determine fuel pin failures as well as gross structural defects in the assemblies.

The first examination was a routine binocular inspection of the fuel assemblies (**12 THP 6040 PER.353). As each assembly is downloaded to the Spent Fuel Pool, it is examined on all four sides visually. The examiner is looking specifically for torn or missing gridstraps, missing or damaged fuel pins, excessive clad hydriding, or rod bow to gap closure. This inspection is primarily intended to detect fuel damage caused by mechanical interaction between assemblies or baffle jetting, and is done during each refueling. There was no indication of any fuel damage.

Due to RCS chemistry levels indicative of several leaking fuel pins, a contract was let to Advanced Nuclear Fuels (ANF) to provide Ultrasonic (UT) examination of the assemblies making up the Unit 2 Cycle 6 core, as well as any replacement assemblies for Cycle 7. the Ultrasonic system works by a probe transceiver sending a high frequency sound wave into a fuel pin and measuring the strength of the returning signal, or "ring back". A fuel pin can be determined to have water in it by monitoring the relative strength of this ring back. In this way, not only can an assembly be determined to have leaking pins, but the numbers and locations of the bad pins can be identified.

Testing results:

Assemblies Tested:	196
Assemblies with Failures	6
Number of Failed Fuel Pins	9

Fuel Batch	Vendor	Assemblies Tested	Assemblies	Pins
M (1 time burned)	W	3	0	0
R (3 times burned)	W	1	0	0
S (3 times burned)	ANF	12	0	0
T (2 times burned)	ANF	92	5	8
U (1 time burned)	ANF	88	1	1
		196	6	9

Three of the failed fuel pins (2 in assembly T24 and 1 in assembly U39) were on the periphery. Using the small camera mounted on the testing system, a short video inspection was performed. The rods in assembly T24 both showed signs of secondary hydriding. The rodlet in assembly U39 was inconclusive.

Indiana Michigan
Power Company
Cook Nuclear Plant
P.O. Box 458
Bridgman, MI 49106
616 465 5901

AEP
INDIANA
MICHIGAN
POWER

February 28, 1989

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Donald C. Cook Nuclear Plant
Docket Nos. 50-315/50-316
License Nos. DPR-58/DPR-74

Document Control Manager:

Two copies of the 1988 Annual Operating Report for the Donald C. Cook Nuclear Plant are being transmitted to you under this cover letter. The information contained in this report covers the activities delineated in the Donald C. Cook Nuclear Plant Technical Specifications, Section 6.9.1.5, and the requirements of 10 CFR 50.59.

Copies of this report have been transmitted to the Regional Administrator, the Director of Inspection and Enforcement, the Director, Office of Management Information and Program Control of the United States Nuclear Regulatory Commission and the NRC Resident Inspector as specified in 10 CFR 50.4 and 10 CFR 50.59.

Respectfully,

W.G. Smith, Jr.
Plant Manager

cc: D.H. Williams, Jr.
M.P. Alexich
P.A. Barrett
S.J. Brewer
J.F. Kurgan
A.B. Davis, Regional Administrator, Region III
Director, Inspection and Enforcement
Director, Office of Management Information
and Program Analysis
NRC Resident Inspector, Bridgman
G. Charnoff, Esq.
R.C. Callen, MPSC
D. Hahn
INPO
Dottie Sherman, ANI Library

Ltr Encl
IE47 Add: NMSS/IMSB 1 1
RES Brooks, B 1 1
1/1