DONALD C. COOK NUCLEAR PLANT

ANNUAL OPERATING REPORT

1979

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INTRODUCTION

The D. C. Cook Nuclear Plant, owned by the Indiana & Michigan Electric Company and located five miles north of Bridgman, Michigan consists of two 1100 MWe pressurized water reactors. The nuclear steam supply systems for both units are supplied by Westinghouse with a General Electric turbine-generator on Unit 1 and a Brown-Boveri turbine-generator on Unit 2. The condenser cooling method is open cycle, using Lake Michigan water as the condenser cooling source. The D. C. Cook Nuclear Plant is the first nuclear facility to use the ice condenser reactor containment system, which utilizes a heat sink of borated ice in a cold storage compartment located inside the containment. The architect/engineer and constructor was the American Electric Power Service Corporation.

This report was compiled by Mr. R. D. Begor, with information contributed by the following individuals:

D. C. Palmer	-	Personnel Exposure Summary
H. Bolinger	-	Inservice Inspection
R. S. Keith and E. A. Abshagen	-	Changes to Facility

PERSONNEL EXPOSURE SUMMARY

The following table represents a tabulation on an annual basis of the number of plant, utility and other personnel receiving exposure greater than 100 Mrem/year and their associated man-rem exposure according to work and job functions.

Assignment of personnel to various groupings is based on what type of work they are usually involved with. Specifically, assignments are made as follows:

Maintenance Personnel -- Includes non-exempt (non-supervisory) personnel from the Maintenance Department and from the Control & Instrument Section of the Technical Department.

Operating Personnel -- Includes non-exempt personnel from the Operations Department, from the Chemical Section of the Technical Department, from the Quality Assurance Department and Security Personnel.

Health Physics Personnel -- Includes non-exempt personnel from the Radiation Protection Section of the Technical Department.

Supervisory Personnel -- Includes exempt (supervisory) personnel from all departments who function primarily as supervisors of non-exempt personnel.

Engineering Personnel -- Includes personnel not primarily functioning as supervisors of non-exempt personnel. This includes such personnel as maintenance engineers, nuclear engineers, performance engineers and station management.

REPORT OF NUMBER OF PERSONNEL AND MAN-REM BY WORK AND JOB FUNCTION
1979

WORK & JOB FUNCTION	NUMBER OF PERSONNEL (> 100 m/Rem) Station Employees	Utility Employees	Contract Workers and Others	TOTAL MAN-REM Station Employees	Utility Employees	Contract Workers and Others
REACTOR OPERATIONS & SURVEILLANCE						
Maintenance Personnel	59	0	39	3.168	0	2.159
Operating Personnel	65	0	0	33.171	0	0
Health Physics Personnel	10	0	5	1.161	0	0.523
Supervisory Personnel	12	4	1	2.232	0.106	0.015
Engineering Personnel	6	0	1	0.971	0	0.105
ROUTINE MAINTENANCE						
Maintenance Personnel	98	0	171	91.298	0	25.002
Operating Personnel	7	0	0	0.521	0	0
Health Physics Personnel	13	0	17	4.296	0	2.592
Supervisory Personnel	8	2	1	2.032	0.154	0.100
Engineering Personnel	6	1	0	0.228	0.138	0
INSERVICE INSPECTION						
Maintenance Personnel	68	0	232	15.841	0	78.863
Operating Personnel	9	0	0	1.244	0	0
Health Physics Personnel	5	0	13	0.559	0	6.254
Supervisory Personnel	8	7	7	3.025	1.001	4.493
Engineering Personnel	8	1	0	0.585	0.107	0
SPECIAL MAINTENANCE						
Maintenance Personnel	84	0	448	28.626	0	171.743
Operating Personnel	2	0	0	0.063	0	0
Health Physics Personnel	10	0	18	4.131	0	6.248
Supervisory Personnel	6	16	18	0.576	12.802	5.379
Engineering Personnel	5	14	5	0.469	5.609	0.960
WASTE PROCESSING						
Maintenance Personnel	53	0	131	12.009	0	57.084
Operating Personnel	4	0	0	0.362	0	0
Health Physics Personnel	12	0	15	6.423	0	2.673
Supervisory Personnel	8	2	8	0.795	0.159	3.043
Engineering Personnel	4	0	0	1.196	0	0
REFUELING						
Maintenance Personnel	70	0	112	13.716	0	53.157
Operating Personnel	1	0	0	0.059	0	0
Health Physics Personnel	6	0	26	0.438	0	12.872
Supervisory Personnel	10	1	6	2.340	0.107	3.823
Engineering Personnel	6	0	1	2.061	0	0.121
TOTAL						
Maintenance Personnel	98	0	642	138.658	0	388.008
Operating Personnel	65	0	0	35.420	0	0
Health Physics Personnel	14	0	34	17.008	0	31.162
Supervisory Personnel	23	17	22	11.000	14.329	13.553
Engineering Personnel	16	17	6	5.510	5.854	1.186
GRAND TOTAL	216	34	704	207.596	20.183	433.909

INSERVICE INSPECTION

In December 1979 an inservice examination of tubing contained within the Unit 2 Steam Generators (No. 2 and No. 3) was conducted by Westinghouse Corporation during Refueling Cycle I-II. This examination was performed to satisfy the surveillance requirements identified in Section 4.5.5 of the Donald C. Cook Nuclear Plant Technical Specifications. The results of this inspection revealed the following:

1) A total of 644 Steam Generator tubes were examined through the "U" bend area. This number is in excess of the 3 percent required by Technical Specification and ASME Code Section XI.

2) This examination revealed no tubing defects having a penetration greater than 20 percent of wall thickness.

3) There were no tubes plugged in any of the Steam Generators.

Evaluation of the inservice eddy current examination data was accomplished by a Zetec, Inc. data interpreter certified to Level IIA using Zetec equipment calibrated in accordance with Zetec procedure.

CHANGES TO FACILITY

Brief descriptions and summary safety evaluations for design changes (RFC's) 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 without prior Nuclear Regulatory Commission approval pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59(a).

DC-12-958

Infrared detectors were installed in certain CO_2 protected areas of the D. C. Cook Nuclear Plant. Additional alarm units for the new detectors were installed in each Control Room.

The control circuitry logic was modified so that both the ionization and the infrared detectors must see a fire condition in order for CO_2 actuation to occur.

This change will minimize false dumps of CO_2 from dust or wind conditions which may exist in these areas.

This RFC is not considered safety related because the changes are being made to Seismic Class III equipment and while applicable are not required by the NRC fire protection recommendations. This RFC will eliminate potential spurious actuations of the CO_2 suppression system and was accounted for in the fire hazards analysis for Cook Plant. This RFC provides an additional level of defense in depth for fire protection at the D. C. Cook Nuclear Plant.

DC-12-1521

RFC-DC-12-1521 changed the fire system actuation method from increasing water pressure to loss of air on the following sprinkler systems served by the Auxiliary Building fire header:

1. Drumming area (preaction type)
2. New fuel receiving area (preaction type)
3. Aux. Bldg. H_2 tubes (dry pilot)

The Auxiliary Building fire header is normally filled with a static head of water. Valves ZMO-10 & 20 isolate the header from the rest of the plant fire suppression water header which is normally pressurized. The system is designed so that upon actuation of any auxiliary building sprinkler system (in the event of a fire) the static head of water in the header would be sufficient to create a 25 psi increasing pressure signal to trip

the deluge valve pressure switch which, through the fire system logic, opens ZMO-10 & 20 and starts a fire pump. The fire system logic would then kick in additional pumps as needed depending on the water demand to extinguish a fire. The results of surveillance testing of this system have shown that the static head does not provide sufficient pressure increase to actuate the deluge valve pressure switches.

Since the auxiliary building sprinkler systems are either the dry pilot or preaction type, air must be vented from the dry pilot actuator in order to open the deluge valves so that a water flow path to the sprinkler heads is established. The air supply is normal plant air. The subject RFC allows the loss of air (decreasing air pressure) to provide the fire system logic input to open ZMO-10 & 20 and start the fire pump. The fire system logic would then function as previously indicated.

The fire protection system is Seismic Class III, however it provides protection to safety related areas and equipment in the auxiliary building and hence this RFC is considered to be safety related. Additionally, the subject RFC closes out an open item of the NRC reviews.

The Nuclear Safety & Licensing Section has reviewed the subject RFC in light of the requirements of NRC Branch Technical Position APCSB 9.5-1 and the AEPSC defense in depth philosophy. The results of this review indicate that this RFC will further enhance the ability of the fire suppression water system to perform its intended design function while in no way degrading the remainder of the fire protection systems at the Donald C. Cook Nuclear Plant.

RFC-DC-12-1521 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59. This RFC will in no way adversely affect the health and safety of the public.

DC-12-2111

A second oxygen analyzer has been installed to continuously sample the discharge line from the Waste Gas Compressors. Upon a signal of high oxygen concentration, automatic control features will isolate the Gas Decay Tank being filled and switch the flow automatically to the "standby" tank. The affected component is then purged with nitrogen to dilute the oxygen concentration. Additional oxygen sampling is completed to verify that the oxygen concentration has decreased to a safe level.

This RFC is considered safety-related since it involves the addition of a second oxygen analyzer to a Class I system and because certain sections of the pipe and valves being added are to be designed and installed as Class I to meet isolation criteria for such systems. The second oxygen analyzer serves the purpose of sampling continuously the discharge line from the

waste gas compressors. It serves to start control operations to avoid the development of explosive mixtures in the gaseous waste disposal system. The installation of the second analyzer was requested in question 320.2 of the FSAR (Appendix Q) with additional provisions being mentioned in question 320.6.

Because the nature of the modification itself, and because of its purpose, this RFC does not constitute an unreviewed safety question as defined in 10CFR50.59 nor will it endanger the health and safety of the public.

DC-12-2115

The thermocouple meter readout for temperature indicator ITI-900 on the Refueling Water Storage Tank (RWST) was replaced with a more accurate Omega digital readout unit for obtaining RWST temperature.

The low temperature alarm setpoint for ITA-900 on the RWST was changed from 40°F to 80°F. This new alarm setpoint will bring the alarm in before Technical Specifications values are exceeded.

These changes were recommended for compatibility with the Emergency Core Cooling System (ECCS) analysis reported in Appendix I to the Final Safety Analysis Report. The new temperature alarm set point is the minimum temperature specified in the Unit 2 Technical Specifications.

These changes are safety-related because the RWST and associated components are Seismic Class I and are part of the ECCS.

This RFC does not constitute an unreviewed safety question as defined in 10CFR50.59.

DC-02-2121

The following modifications were made to the Unit #2 Containment Purge and Exhaust System in accordance with the requirements of the NRC's Branch Technical Position CSB6-5:

1. Installed orifice plates and debris screens on the inboard isolation valves (VCR-101 through VCR-107) at the seven containment purge ventilation penetrations.
2. Provided a tiedown system with blowout panels on the purge system ductwork (exterior to the containment) which could become missiles and damage Class I safety equipment in the area if such ductwork were subjected to LOCA pressures.
3. Modified the 14 isolation valves (VCR-101 through VCR-107 and VCR-201 through VCR-207) to provide valve closure times of less than 5 seconds.

The subject RFC has been initiated to meet the commitments in Appendix Q of the FSAR (Questions 022.4 and 022.13) and requirements of Branch Technical Position CSB-6-4, "Containment Purging During Normal Plant Operation." The changes enhance the reliability of the involved systems and do not prevent the systems from performing their intended safety function. The changes do not constitute an unreviewed question as defined in 10CFR50.59.

DC-12-2158

RFC-DC-12-2158 installed an automatic composite sampler on the turbine room sump discharge line. Installation of this composite sampler allows for the collection and analysis of sump discharge samples during periods of primary-to-secondary steam generator tube leakage.

This request for change (RFC) revises a Seismic Class III system which is not required for the safe shutdown of the unit, and thus is considered to be a non-safety related change. The subject sampler is being installed in accordance with NRC question 320.8 to the FSAR. RFC-DC-12-2158 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59. This RFC provides for design changes which are within the limits of our safety analysis and hence will have no adverse effect on the health and safety of the public.

DC-12-2166 Unit #2 Only

Block and drain valves were added to the following systems on Unit #2 of the D. C. Cook Nuclear Plant:

1. Emergency Core Cooling System.
2. Primary Water System
3. Component Cooling Water System.
4. Ice Condenser Refrigeration System.
5. Demineralized (Makeup) Water System.

These valves were added to facilitate pneumatic testing of seat leakage on the containment isolation valves in these systems. This testing is required by Unit 2 license condition 3.1 in accordance with the committment made in response to Questions 022.7 and 022.15 contained in Appendix Q to the FSAR.

This RFC is safety related because it requires modifications to be made to some containment penetrations which are Seismic Class I and are required to be isolated under certain design basis accident conditions. The modifications are being made in accordance with the design basis for containment isolation as presented in the FSAR. This RFC does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.

DC-02-2181

All Unit #2 Pressurizer Level, Steam Generator Level (narrow range) and Reactor Coolant Pressure (wide range) transmitters were replaced with ITT Barton requalified transmitters to fulfill the commitment made in response to Question 022.8 in Appendix Q to the FSAR.

The change required by this RFC involves the replacement of 17 Class IE signal transmitters. The replacement is the result of the Seismic and Environmental Requalification program conducted by Westinghouse Electric Corporation.

Although this change is safety related, the required seismic analyses have been completed on the new replacement sensors, their supports and tubing. This RFC is not considered to be an unreviewed safety question in accordance with 10CFR50.59 (2) (a), and will not adversely affect the health and safety of the public.

DC-01-2193

On the Unit #1 Emergency Diesel Generators, the control circuitry and the feed breaker control circuitry were modified to electrically remove the "non-essential" trip circuits from service whenever the diesel generator is required for accident conditions (Safety Injection or blackout). The "non-essential" trip circuits remain operable during Surveillance Testing until an emergency signal is received.

During the accident mode the "essential" trip circuits (Engine Overspeed and Generator Differential) will remain in service to trip the diesel should any essential trip condition arise.

The changes incorporated by this RFC are safety related in that the Diesel Generators are Seismic Class I equipment and are served by Class IE cable. These changes do not represent an unreviewed safety question in accordance with 10CFR50.59a(2) and further, they represent an upgrade in safety in that additional protection is afforded by the enhancement of Diesel Generator availability during a Safety Injection Signal. Hence, these changes shall not affect the health and safety of the public.

DC-02-2202

Backup molded case circuit breakers and/or fuses were added for 600V ESS and Non-ESS electrical equipment inside the Unit #2 Containment of the D. C. Cook Nuclear Plant. The addition of redundant circuit breakers will reduce the probability of penetration damage during a fault.

RFC-DC-02-2202 proposes the installation of a back up or redundant set of breakers and/or fuses to protect the 600 volt ESS and Non-ESS electrical equipment in the containment building because failure of a single circuit

breaker to open during a fault may cause the electrical penetration to be damaged.

This RFC is in compliance with "additional condition (m)" to Unit #2 Operating License.

Since the installation of this equipment adds a redundancy for protection of safety related equipment, it does not constitute an unreviewed safety question as defined in 10CFR Section 50.59.

DC-01-2207

All Unit #1 reactor coolant wide range pressure, pressurizer pressure and Steam Generator water level transmitters were replaced with requalified transmitters manufactured by ITT Barton to provide upgraded environmental qualification.

This modification involves Siesmic Class I components. The change is safety related because the services involved are to be relied upon for long term monitoring following a design basis accident. In addition, the steam generator level transmitter provides one of several redundant signals to initiate reactor trip and start-up of the auxiliary feedwater system upon a loss of normal feedwater or feedline break. The NRC was informed of this change in a letter dated May 4, 1979 from Mr. G. P. Maloney to Mr. Harold R. Denton (AEP:NRC:00142). This is the same change in Unit 1 as was made in Unit 2 under RFC-DC-02-2181.

This RFC does not constitute an unreviewed safety question as defined in 10CFR50.59 and will not endanger the health or safety of the public.

DC-12-2222 Unit #2 Only

The source of power to the Turbine Driven Auxiliary Feedwater pumps (TDAFP) discharge valves and trip and throttle valves was changed from AC to DC power. This was accomplished by the addition of an "N" train battery as the new power supply for these valves.

The 250 V DC "N" train battery system consists of one battery (one set of 120 lead acid cells); two battery chargers, each supplied from a separate safety train a-c bus; and two standby circuits from the existing AB and CD plant batteries. This "N" battery is physically and electrically isolated from the other plant batteries. Like the other plant batteries, it will have its own active normal charger and a wired standby charger.

The auxiliary feedwater to steam generator valves are normally open; therefore, in most cases, they will not be a load on the battery, but if they (or any among them) happen to be closed the battery has adequate capacity

to drive them open. The remaining load consists of the auxiliary feedwater turbine control bus. The AFW turbine control bus encompasses the AFW turbine start and trip circuits, the overspeed monitor, the test valve, and the emergency leak-off valve. The battery is sized to allow anticipated operation of the valves and their control circuits with the battery chargers and backup feed circuit open. The battery will be capable of serving the turbine driven auxiliary feedpump for as long as the steam supply to the turbine is available. The "N" train battery is further described in Section 8.3.5 on page 8.3.8 of the FSAR.

The NRC required that the change from AC to DC power be made since their generic studies show that the auxiliary feedwater system is too dependent on AC power (offsite & emergency diesel). The reliability of the system will be increased by adding diversity to the power supply such as changing to DC power. This requirement had been imposed by the NRC during the Operating License review for D. C. Cook Unit 2 and license condition 3.K requires that this change be completed prior to startup from the first refueling on Unit 2. This RFC is considered safety related because the Auxiliary Feedwater System is Seismic Class I and the associated electrical hardware is Class IE. Also, the Auxiliary Feedwater System is required to function during a design base accident.

The Nuclear Safety & Licensing Section has reviewed the engineering and design work required to affect this change on both Units 1 & 2, and finds it acceptable for installation.

RFC-DC-12-2222 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59. This change will not adversely affect the health and safety of the public.

<u>DC-12-2230</u>

Additional ionization type fire detectors were installed in the following locations in the Auxiliary Building and in both the Unit #1 and Unit #2 Control Rooms:

1. Spray additive tank room.
2. Nuclear Sampling Room.
3. Reciprocating and centrifugal charging pump rooms.
4. Safety injection pump rooms.
5. HVAC mezzanine above access control area.
6. Laundry room.
7. HVAC vestibules on 633' elevation.
8. HVAC rooms above the control rooms.
9. Above control room ceilings.
10. In control room supply/return air ducts.
11. Unit 1 and 2 hot shutdown panels.

This additional detection capability was required due to the fire hazards and fire loading in these safety-related areas of the Cook Nuclear Plant. The ionization detectors will provide early warning of the signs of a fire thereby allowing the fire brigade sufficient time to proceed to the affected area while the fire is in its incipient stage.

This RFC is considered safety related because the fire hazards in these areas involve Class IE cabling and/or Seismic Class I equipment. The Nuclear Safety & Licensing Section has reviewed this RFC in light of the NRC B.T.P. 9.5-1 and the Fire Hazards Analyses. The results of this review indicate that this RFC will increase the fire detection capability and is consistent with our commitments to the NRC.

RFC-DC-12-2230 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.

DC-12-2225 Unit #2 Only

A Reactor Coolant Pump Motor oil spillage protection and control system was installed on all 4 reactor coolant pump motors of Unit #2.

The oil spillage protection and control system consists of a package of splash guards, catch basins, and enclosures assembled as attachments to the RCP motor at strategic locations to preclude the possibility of oil making contact with hot RCS components and piping.

The oil spillage protection and control system includes an oil-tight enclosure around the high-pressure oil lift system and a set of drip pans, splash guards, and catch basins around the motor lower bracket and bearing, external heat exchanger, and the upper bearing oil reservoir alarm housing.

This system is designed to control both pressure and gravity type oil leaks thus minimizing the possibility of oil ignition from hot reactor coolant piping and other sources.

Each Reactor Coolant Pump (RCP) motor contains a 265 gallon lubricating oil reservoir coupled to the RCP oil lift system which is necessary for the proper operation of the pump. The RCP's are Seismic Class I. While not required for safe shutdown of the plant nor any ECCS functions, the RCP's are part of the Reactor Coolant System pressure boundary. Thus, this RFC is considered to be safety related.

The fire hazards analysis showed that the quantity of lube oil represented a significant fire hazard. The potential for a fire is further increased since the hot RCS piping could ignite the oil should a leak occur. The ignition temperature of the oil is in the same range as the RCS piping temperatures. The existing drip pans were shown to be sufficient to contain

ordinary drip oil however, the oil lift system is pressurized and a pressurized oil leak could not be handled. Also in light of the fact that a heavy accumulation of electrical cable trays are in the vicinity of each pump, a fire involving a pressurized oil leak could potentially have safety significance.

The Nuclear Safety & Licensing Section has reviewed this RFC in light of NRC Branch Technical position APCSB 9.5-1 and the Fire Hazards Analysis. This review indicates that this change increases the fire protection capability in the Cook Nuclear Plant while not, in any way, degrading any safety related system. The Westinghouse scope includes the appropriate seismic, missile and high energy line break (LOCA and oil jet) analyses.

RFC-DC-12-2225 does not create a substantial safety hazard nor does it constitute an unreviewd safety question as defined in the 10CFR50.59. This RFC further enhances the fire protection systems in the Cook Nuclear Plant.

DC-12-2231

Automatic fire protection water sprinkler systems were installed in the following areas of the D. C. Cook Nuclear Plant:

a) Diesel engine driven fire pump rooms.
b) Charging and Safety Injection pump rooms.
c) Auxiliary Building Elev. 587' East End.
d) Auxiliary Building Elev. 587' West End.
e) Auxiliary Building Elev. 609' Laundry Room area.
f) Reactor Coolant Pumps in both Unit #1 and Unit #2 Containments.

These sprinkler systems are necessary to protect safety related equipment/cabling from the effects of a postulated fire either due to the permanently installed combustible materials or transient fire loads in the above areas. Thus, this RFC is considered to be safety related.

The Nuclear Safety & Licensing section has reviewed the subject RFC in light of the requirements of NRC B.T.P. 9.5-1 and the Fire Hazards Analysis. This review indicates that these sprinkler systems will improve the automatic fire extinguishing capability in the Cook Nuclear Plant and meet commitments to the NRC. In addition, the sprinkler system at the Reactor Coolant Pumps (RCP) were installed in conjunction with RFC DC-12-2225, which provides the RCP Motor Oil Spillage Control/Protection System, and in this manner we are increasing our capability to extinguish a fire involving lube oil should one occur. This is an added level of protection for fighting in-containment fires.

RFC DC-12-2231 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59. This RFC will not adversely affect the health and safety of the public.

DC-12-2276 Unit #2 Only

Local Control capabilities were provided for the Unit #2 Emergency Diesel Generators of the D. C. Cook Nuclear Plant. This modification provides for: 1) Starting, stopping, controlling speed and voltage, and starting required auxiliaries from a location other than the Control Room; 2) Isolating the existing Diesel Generator controls in the Control Room; and 3) Closing the Diesel Generator breakers locally.

A new sub-panel DGABX (DGCDX) was installed in each Diesel Generator room, with similar controls and instrumentation as the control room. No start-stop capability was built into the new panel because the diesel can be started and stopped from the existing sub-panel DGAB (DGCD) located in the Diesel Generator room. The speed and voltage controls as well as monitoring instrumentation are located on the new sub-panel DGABX (DGCDX). Also, on the new sub-panel there is a LOCAL/REMOTE transfer switch, to transfer the voltage and speed control from the control room to the new sub-panel. An annunciator will inform the operator in the control room that the Diesel Generator is controlled locally. The instrumentation is not affected by the transfer switch and is operational anytime the diesel is running. During plant normal operation this LOC/REM transfer switch is placed in the remote position and the annunciator is cleared.

This RFC is considered safety-related because electrical circuits being modified are Class IE equipment and the diesel generators are required to function following a loss of offsite power.

One of the assumptions made by AEPSC in the design of the local shutdown system was that offsite power was available to energize the safety buses. A concurrent loss of functionability from the control room (cable vault fire) and loss of offsite power was not part of the local shutdown design basis. Thus local control of the Diesel Generators was not required.

The NRC in License Condition C.3.0.C of the Unit #2 Operating License required that loss of offsite power be included in the design basis which requires provisions for local control of the Diesel Generators.

RFC-DC-12-2276 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.

DC-12-2352

All Unit #1 and Unit #2 Main Steam Flow and Pressurizer Pressure narrow range transmitters were replaced with requalified ITT Barton Transmitters.

Condition 4.A. to the Unit No. 2 Operating License, as amended on June 16, 1978 (Amendment No. 6), required the replacement of all electronic

Foxboro transmitters (E11GM and E13DM) used in safety-related circuits inside containment with transmitters qualified by sequential environmental testing in accordance with IEEE 323-1971. By letter dated June 1, 1978, AEP:NRC:00021, AEP committed to replacing the Foxboro transmitters used in the "Steam Flow" and "Pressurizer Pressure" function on Unit No. 1 with qualified transmitters.

The replacement transmitters are Barton Model No. 763 pressure transmitters (for use in the "Pressurizer Pressure" functional circuits) and Barton Model No. 764 differential pressure transmitters (for use in "Steam Flow" functional circuits). These Barton transmitter Models have been tested in accordance with IEEE 323-1971 and the results of the testing documented in Westinghouse letter NS-TMA-1950. (See AEP:NRC:00095 submittal for details of the tests and test results.)

The "Pressurizer Pressure" and "Steam Flow" functions are not part of the Long Term - Post Accident Monitoring (LT-PAM) Systems (as defined in Technical Specification Table 3.3-10 (Unit No. 2). The Barton Model Nos. 763 and 764 transmitters are fully qualified for their respective reactor trip and/or ESF actuation functions (short term qualification.)

Based on the above, NS&L has no reason to object to the installation of the Barton transmitters. The subject RFC does not represent an unreviewed safety question as defined in 10CFR50.59 and implementation of this change will have no adverse effect on the health and safety of the general public.

<u>DC-01-2353</u>

The eight part-length control rods were removed from the Unit #1 reactor. An anti-rotational device was installed on each part-length control rod drive mechanism (CRDM) to prevent the CRDM lead screws from rotating in the direction which would lower them due to gravity or vibration. Thimble plugs were installed in place of each part-length control rod to prevent any thermal or hydraulic problems associated with the part-length removal.

The Part Length Rods on Donald C. Cook Unit No. 1 will be removed based on the following:

1. No credit is taken in the Safety Analysis performed by Exxon and Westinghouse for their presence.
2. The reactor's Operating License and Technical Specifications pre-empt their use.
3. Unit 1 has successfully load-followed and controlled artificially large Xenon oscillations without the use of the Part Length Rods in Cycles 1 and 2.
4. The Part Length Rods have been removed from Unit No. 2.
5. The rods will be stored in the spent fuel pool should future changes in licensing requirements or operating considerations permit their use.

The FSAR will be amended to eliminate references to the use of part-length control rods in Unit 1, as was done for Unit 2 in Amendment 78. A Technical Specification change request will be submitted to the NRC to eliminate Technical Specification 3/4.1.3.6 which requires that all the part-length control rods be fully withdrawn in Modes 1 and 2.

This RFC does not constitute an unreviewed safety question as defined in 10CFR50.59 and will not endanger the health or safety of the public.

DC-01-2354

The following modifications were made to the Unit #1 Containment Purge and Exhaust System:

1. Installed debris screens on the inboard isolation valves at the seven containment purge ventilation penetrations.

2. Provided a tiedown system with blowout panels on the purge system ductwork exterior to the containment which could become missiles and damage Class I safety equipment in the area if such ductwork were subjected to LOCA pressures.

3. Installed quick release valves in the air lines at those containment purge isolation valves which do not meet a 5-second closure time following a containment isolation signal.

RFC DC-01-2354 calls for modifications to the Unit 1 Containment Purge Supply and Exhaust System to comply with the NRC's Branch Technical Position CSB 6-4. These modifications were made to Unit 2 as indicated in our response to Question 022.4 and 022.13 in Appendix Q to the FSAR. The generic NRC letter of November 28, 1978 required that all operating plants who desire to continue purging be modified to comply with B.T.P. CSB 6-4 or justify otherwise. The NRC is concerned about the ability of the Purge System isolation valves to close under the dynamic forces associated with a LOCA. The Purge System isolation valves are Seismic Class I and are required to function during a design basis accident. As such, this RFC is considered to be safety related.

The Nuclear Safety & Licensing Section has reviewed this RFC in light of our past NRC correspondence and meetings on this matter. This RFC is consistent with the commitments made in our January 4, 1979 letter to the NRC (AEP:NRC:00114).

RFC DC-01-2354 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10 CFR 50.59.

DC-02-2355

The temporary anti-rotational devices on the part length control rod device mechanism (CRDM) lead screws were replaced with permanent devices on Unit #2 of the D. C. Cook Nuclear Plant.

Plans to install the permanent devices were reported to the NRC in letters AEP:NRC:00117 and AEP:NRC:00190 concerning FSAR Amendment No. 83. The installation of the permanent devices is expected to save considerable outage time and human radiation exposure due to elimination of the need for periodic inspections.

This RFC is safety related because control rod assemblies are Seismic Class I, have the potential for interfering with other Siesmic Class I systems, and are described in the FSAR. The stress analysis is to be provided by Westinghouse.

This RFC does not constitute an unreviewed safety question as defined in 10CFR50.59 and will not endanger the health or safety of the public.

DC-12-2361

The following modifications were made to the Containment Recirculation Sump in both Units of the D. C. Cook Nuclear Plant;

1. Removal of the perforated plate below the sump outlet pipes. (Blockage of this plate was shown to produce swirl which may be undesirable.)

2. Extension of the existing vent pipe to an elevation above the maximum level possible in the containment during an accident.

3. Modification of the secondary sump roof to incorporate a sloped roof to allow more effective venting of the primary sump.

4. Boring vent holes in the top cover of the primary sump upstream of the cranewall to allow air to escape from beneath this cover.

5. Modification of the sump inlet to incorporate both coarse and fine screens.

These modifications are addressed in an Alden Research Laboratory (ARL) report, "Hydraulic Model Investigation of Vortexing and Swirl Within a Reactor Containment Recirculation Sump." In addition, a commitment was made to perform these modifications during the first and fourth Refueling Outage on Unit #2 and Unit #1 respectively (AEP:NRC:00110).

This RFC calls for various structural modifications and additions to the containment recirculation sump. Since the sump is an integral component of the Emergency Core Coolant System (ECCS) and as such is Seismic Class I, the RFC was deemed to be safety related. Even without the proposed modifications, the sump was shown to perform acceptably as demonstrated by model testing performed by Alden Research Laboratory (ARL). However, ARL did suggest certain changes which would further improve the sump performance and those recommendations are the basis for the RFC. These changes will improve the already satisfactory performance of a safety function and as such do not constitite an unreviewed safety question as defined in 10CFR50.59.

DC-12-2385

A "unit trip" upon loss of both main feedwater pumps was installed on both Units of the D. C. Cook Nuclear Plant. A signal derived from the feed pump turbine stop valve closure trip circuit (loss of feed pump) was provided to trip the Main Turbine directly. A Turbine Trip above 10% power (reactor and turbine power) will also result in a trip of the Reactor. This new trip signal was added to provide additional margin in the Steam Generator secondary side water inventory in the event of a loss of the main feedwater pumps.

The circuit involved is not Class IE and the mechanical equipment involved is Seismic Class III. The Reactor Trip circuit (above 10% power) derived from a Turbine Trip, and vice versa, already exists. The direct Reactor Trip signals which can be derived from a loss of secondary heat sink transients are:

1. Low-Low Steam Generator water level.

2. Low Steam Generator water level in coincidence with steam flow/feed flow mismatch.

3. Pressurizer Pressure - high.

4. Overtemperature ΔT.

5. Low Steam line pressure.

6. Containment Pressure - high.

The loss (or partial loss) of secondary heat sink transients which are analyzed in the FSAR are the loss of normal feedwater, loss of load, load rejection, turbine trip, main steam isolation valve closure, feedwater system malfunction, main feedwater line break accident, and main steam line break accident. Any one of the above reactor trip signals meets the requirements of the safety analysis and are required to be operable by our Technical Specifications. The safety analysis shows that for a loss of normal feedwater (main feed pump trip) there is adequate

secondary side water inventory to prevent unacceptable safety related consequences and that the auxiliary feedwater system provides the necessary heat sink to remove decay heat.

The addition of this new trip signal should provide added margin to the already adequate secondary water inventory during a main feed pump trip transient and as such there is no safety analysis requirement for this trip to occur (no credit is taken for it). Low-low steam generator water level or low steam generator water level coincident with steam flow/feed flow mismatch typically indicate the loss of main feed pumps and provide adequate trip signals as part of the Reactor Protection System. This new trip circuit being installed under RFC-DC-12-2385 is not part of the Reactor Protection System, however, it does, in effect, provide an additional Reactor Trip Signal when the reactor and turbine power are above 10% and as such RFC-DC-12-2385 is considered to be safety related.

The Nuclear Safety and Licensing Section has also reviewed RFC-DC-12-2385 in light of the recent analysis performed as a result of the TMI-2 accident. This analysis shows that in the unlikely event of a loss of main feedwater and the simultaneous loss of auxiliary feedwater with the reactor initially at 100% power, the secondary side water inventory would boil off in approximately 42 to 45 minutes. A reactor trip would occur in approximately 11 to 13 seconds on SF/FF mismatch coincident with low S/G water level. This long period of time for boil off is attributed to the large secondary side water inventory which exists in the steam generator. RFC-DC-12-2385 has no significant effect on mitigating the consequences of such a transient at Cook Plant and at best will result in an earlier Turbine/Reactor trip. If this new trip circuit did not function, the safety analysis requirements are met with the Reactor Protection trip circuits described above. As such, the effect of RFC-DC-12-2385 on the consequences of loss of normal feedwater transients is in the conservative direction.

The Nuclear Safety and Licensing Section has no reason to object to the installation of this new trip circuit under RFC-DC-12-2385. Tripping the reactor sooner for a loss of feed pump event is in the conservative direction by preserving some additional secondary side water inventory for this particular transient. This positive aspect leads us to conclude that RFC-DC-12-2385 has no negative effect on plant safety.

RFC-DC-12-2385 does not create a substantial safety hazard nor it constitutes an unreviewed safety question as defined in 10CFR50.59. This RFC will not, in any way, adversely effect the health and safety of the public.

<u>DC-12-2387</u>

The Safety Injection actuation logic was modified to remove "low pressurizer water level" as a coincident parameter with "low pressurizer pressure." Safety Injection will now be actuated by a 2 out of 3 signal from pressurizer pressure only.

This modification assures safety injection actuation for Pressurizer steam space breaks and other small break LOCA's. The new actuation method is bounded by the current accident analyses contained in the FSAR. A two out of three logic is necessary to avoid spurious actuations of safety injection in accordance with applicable criteria. The above modifications to the logic were required by the NRC in action item 3 of IE Bulletin 79-06A, Revision 1 following the TMI-2 accident. License Amendment No. 29 for Unit 1, and Amendment No. 11 for Unit 2 were issued by the NRC and incorporated the logic changes made under this RFC into the Cook Plant Technical Specifications. This RFC is safety related because the logic modifications are being performed on the RPS/ESFAS (Reactor Protection System/Engineered Safety Features Actuation System) circuits which are Class IE. This RFC does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.

DC-12-2392

The Emergency Diesel Generator breaker control circuitry was modified to assure breaker closing. If a loss of normal power occurs while one Diesel Generator output breaker is locked out, the Diesel Generator Engine will automatically start. After sensing the 4KV loss of voltage on the opposite bus, the breaker which was locked out will close into the dead bus automatically after the control switch is removed from lock-out and placed into the "after trip" or neutral position.

In the event that the breaker control switch is in the lock-out position (this condition is alarmed), these circuit modifications will allow the operator to close the breaker, thereby energizing the ESF buses with the diesel generator running. This modification eliminates the need for the operator to stop the diesel, close the breaker and restart the diesel under this condition. This RFC adds an additional level of defense for manual control of the diesel generator circuits and is not required by the NRC. This RFC is safety related because modifications are to be made to the diesel generator circuits which are Class IE. This RFC does not create a substantial safety hazard nor does it constitite an unreviewed safety question as defined in 10CFR50.59.

DC-12-2395

RFC-DC-12-2395 modified the Reactor Protection System Safeguards actuation and reset circuit. The subject RFC includes:

1. Providing sealed covers which block operator access to the reset switch.
2. Alarming the reset condition of any safeguards output signal whether or not automatic safety actuation signals are blocked by this action.
3. Hard wire a trip close signal from Containment isolation - Phase A directly to each Containment purge supply and exhaust isolation valve such that a Containment Ventilation Isolation signal can be overridden, the purge valves re-opened, and a subsequent Phase A signal will automatically trip the valves closed.

This modification was installed at the NRC's request to prevent manually overriding containment isolation signals to allow continued purging with an isolation signal present.

This RFC is considered to be safety related because the safeguard actuation circuits are Class IE Equipment and are required to function under a design basis accident. These modifications are consistent with the design basis of the safeguards actuation circuits for the Cook Plant. As such, the NS&L Section has no reason to object the modifications being installed under the subject RFC.

RFC-DC-12-2395 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59. This RFC will, in fact, prevent inadvertant over-riding of safety actuation signals and will not have any adverse effect on the health and safety of the public.

<u>DC-12-2406</u>

The Steam Generator Lo-Lo Level Reactor Trip/Auxiliary Feedwater pump start setpoints were revised from 11% to 15% in Unit #1 and from 17% to 21% in Unit #2.

Review of the FSAR Steam Line Break Analysis for the D. C. Cook Plant, which bounds the main feedline break, shows the containment temperature increases to less than 330°F. However, the containment temperature never exceeds 200°F prior to receiving a containment Region 1 high pressure trip which produces a reactor trip and auxiliary feedwater actuation. A containment temperature of 200°F corresponds to a correction of 4% in Steam Generator Level.

This RFC calls for changing the setpoints for a reactor trip from the steam generator water level low-low logic channels. This modification is necessary in order to properly account for temperature effects in the reference leg during a design basis accident. The non-conservative aspect of the water level settings were reported generically to the NRC by Westinghouse in accordance with 10 CFR 21. A Cook specific review has determined that the water level bias indicated by Westinghouse, although applicable, did not create a substantial safety hazard nor did it constitute an unreviewed safety question. Correspondingly this RFC adds safety margin to the Cook Plant setpoints and does not constitute an unreviewed safety question as defined in 10CFR50.59.

DC-12-2440

An acoustical monitoring system was installed on both Units of the D. C. Cook Nuclear Plant to provide direct and reliable indication of the position of the Pressurizer PORV's and safety valves.

The system consists of four accelerometers as sensing elements and an LED display as an output monitor. The LED display is mounted in the rod control panel in the Control Room.

Each pressurizer relief valve (SV-45A,B,C) has an accelerometer strapped to its discharge pipe. The fourth accelerometer is strapped to the common header at the discharge of the PORV's. The Control Room display has four separate channels, one for each accelerometer. Each output channel has a series of 10 LED display lights. This vertical display of lights is sequenced to come on in steps indicating relative valve flow.

This modification is required for compliance with recommendation 2.1.3.a of NUREG-0578 entitled, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." The NRC's letter September 13, 1979 as supplemented, required implemented of this "Lessons Learned" recommendation.

The acoustic monitoring system will indicate whether or not there is any flow through these valves thereby giving the operator an early indication of a possible stuck open valve. Since these valves are part of the reactor coolant system pressure boundary, a stuck open valve is equivalent to a small loss-of-coolant accident. NUREG-0578 requires that the operator take appropriate actions to isolate the relief path, if possible, and terminate RCS depressurization. Hence a reliable and direct indication is needed. This RFC is safety related because the NRC requires it to be a safety grade, Class IE, Seismic Class I acoustic monitoring system. This RFC is being installed in accordance with and similar to the applicable criteria as presented in the FSAR. This RFC does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.

DC-12-2450

The existing containment pressure transmitters were respanned to increase the range for monitoring of containment pressures. The lower containment pressure transmitters PPP-300, 301, 302 and 303 were respanned to -5 to +12 psig, (previously span had been -1 to +15 psig). The lower containment pressure trip setpoints remain the same only the transmitter span and indicator scales were changed. Also respanned were the upper containment wide range pressure transmitters, PPA-310 and 312, to -5 to +36 psig (from previous -1 to +12 psig).

This modification was implemented in accordance with the requirements of NUREG-0578, "Lessons Learned" (ACRS Item "Containment Pressure Indication") which required that the sensing instrumentation be capable of monitoring a pressure range of -5 psig to 3X containment design pressure.

This RFC is safety related because it involves changes to Seismic Class I equipment whose circuits are Class IE and is in compliance with the NRC requirement that these indications be safety grade. This RFC does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.

DC-12-2452

An additional alarm was installed to indicate Low-Low water level in the Condensate Storage Tank. This alarm will annunciate when the remaining level in the tank can only supply enough water for 20 minutes operation of the auxiliary feedwater pumps.

This installation fulfills a commitment made to the NRC (AEP:NRC:00300) to implement the Low-Low level alarm by January 1, 1980 as required by NUREG-0578.

This RFC is not safety related since it only serves to alert the operator to this condition. This RFC adds an additional level of defense-in-depth to the engineered safety features in the Cook Plant. This RFC does not create substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10CFR50.59.