DONALD C. COOK NUCLEAR PLANT

ANNUAL OPERATING REPORT

1980

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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 was 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. H. Bolinger	-	Inservice Inspection
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
1980

WORK AND JOB FUNCTION	NUMBER OF PERSONNEL (>100 mRem)			TOTAL MAN-REM		
	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT WORKERS AND OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT WORKERS AND OTHERS
REACTOR OPERATIONS & SURVEILLANCE						
Maintenance Personnel	60	0	33	3.460	0	2.241
Operations Personnel	63	0	0	27.557	0	0
Health Physics Personnel	13	0	11	3.060	0	4.040
Supervisory Personnel	11	0	4	1.020	0	.100
Engineering Personnel	5	3	1	.620	.080	.050
ROUTINE MAINTENANCE						
Maintenance Personnel	96	0	119	54.130	0	12.796
Operations Personnel	1	0	0	.060	0	0
Health Physics Personnel	13	0	20	2.140	0	3.360
Supervisory Personnel	11	0	11	1.540	0	.818
Engineering Personnel	6	4	1	.320	.640	.020
INSERVICE INSPECTION						
Maintenance Personnel	46	0	94	5.250	0	13.292
Operations Personnel	0	0	0	0	0	0
Health Physics Personnel	9	0	10	.810	0	2.770
Supervisory Personnel	5	0	6	.290	0	1.590
Engineering Personnel	6	3	7	.790	.930	2.070
SPECIAL MAINTENANCE						
Maintenance Personnel	80	0	421	21.380	0	162.742
Operations Personnel	1	0	0	.020	0	0
Health Physics Personnel	9	0	22	1.300	0	8.330
Supervisory Personnel	8	0	43	.540	0	15.180
Engineering Personnel	5	12	2	.710	3.120	.120
WASTE PROCESSING						
Maintenance Personnel	52	0	95	7.430	0	37.877
Operations Personnel	25	0	0	3.130	0	0
Health Physics Personnel	13	0	6	6.200	0	1.040
Supervisory Personnel	7	0	5	1.710	0	3.030
Engineering Personnel	2	1	0	2.170	.120	0
REFUELING						
Maintenance Personnel	31	0	50	3.670	0	20.345
Operations Personnel	0	0	0	0	0	0
Health Physics Personnel	1	0	11	.070	0	1.205
Supervisory Personnel	4	0	6	1.020	0	1.750
Engineering Personnel	1	2	1	.150	.140	.030
TOTAL						
MAINTENANCE PERSONNEL	103	0	495	95.320	0	249.293
OPERATIONS PERSONNEL	63	0	0	30.767	0	0
HEALTH PHYSICS PERSONNEL	14	0	25	13.580	0	20.745
SUPERVISORY PERSONNEL	14	0	45	6.120	0	22.468
ENGINEERING PERSONNEL	8	13	8	4.760	5.030	2.290
GRAND TOTAL	202	13	573	150.547	5.030	294.796

INSERVICE INSPECTION

In June and July, 1980 an inservice examination of tubing contained within the Unit #1 Steam Generator No. 3 was conducted by Westinghouse Corporation Nuclear Service Division using a multi-frequency Eddy Current Testing technique. This examination was not required by the ASME Code Section XI 1974 Edition 1975 Summer Addenda, Regulatory Guide 1.83 or Unit #1 Technical Specifications Section 4.4.5.3.a which allowed the inspection interval to be extended to a maximum of once per forty (40) months; nonetheless, the evaluation of the data was in accordance with criteria set forth in the ASME Boiler and Pressure Vessel Code. The evaluation of the examination data was effected by a Zetec, Inc. interpreter certified to Level IIA using Zetec equipment calibrated in accordance with Zetec procedures. The extent and results of this inspection was as follows:

1) Inspected 20 tubes through the seventh support.
2) Inspected 90 tubes, Row 1 through the U-Bends.
3) Inspected 217 tubes through U-Bends other than Row 1.
4) Inspected 380 tubes through the first support.
5) This examination revealed no tubing degradation having a penetration greater than 20 percent of wall thickness.
6) No tubes were plugged.
7) All testing was done from the hot side.

In October, 1980 an unscheduled outage removing Unit #2 from service evolved as a result of a fault in the Main Electrical Generator. When it became apparent that the repairs needed would be extensive, Plant Management and Senior American Electric Power Service Corporation Management conferred and decided to develop a program capable of detecting a very small leak in one of Unit #2 Steam Generators uniquely identified by the Plant as Steam Generator #1.

The following summarizes the techniques, findings and remedial measures taken:

1) The first test program entailed draining and drying the primary side of Steam Generator #1, filling the secondary side with water, the emplacement of a remote operated television camera alternately in the hot and cold leg manways, and the pressurization of the secondary side using nitrogen at pressures ranging from 0 psig to 900 psig. During pressurization, the tube openings were scanned and a visual inspection was made as the camera was moved alternately from the hot leg to the cold leg for evidence of water exuding from a tube(s).

   After several days of monitoring this test no wetness was observed and the test was cancelled and deemed unsuccessful.

2) The second test program consisted of employing Westinghouse Corporation Nuclear Services Division to conduct a multi-frequency Eddy Current Test as directed by Indiana & Michigan Electric Company management. Westinghouse was directed to begin testing by examining all 94 Row 1 tubes over the U-Bend. The program was then expanded using a 4 x 4 pattern to include 223 tubes (9 of which were Row 1 repeats) to the first support plate and 4 tubes full length.

The data obtained was evaluated by a Zetec, Inc. interpreter certified to Level IIA using Zetec equipment calibrated in accordance with Zetec procedures. His interpretation indicated that 4 tubes in Row 1 designated Columns 1, 2, 3 and 4 showed 50 percent degradation and tube designated Row 1 Column 5 showed 20 percent degradation, with all tube wall thinning occurring approximately 15 inches above the tube sheet. Based on this finding, the hand hole covers leading to the Tube Lane Blocking Devices were removed to permit a visual examination of both the tubes and the tube lane blocking device. This inspection revealed that the tube lane blocking device had oscillated, abrading the 5 Row 1 tubes on both the hot and cold leg sides.

Based on this finding, the hand hole covers were removed from the remaining Unit #2 Steam Generators to permit a visual inspection of both the tubes and the tube lane blocking devices. All of these tube lane blocking devices were rigid with the exception of one in Steam Generator #4 which could be moved infinitesimally. There was no evidence of abrasion to any tubes in the area of the tube lane blocking device except as earlier noted in Steam Generator #1.

Based on the findings of the Eddy Current Testing and the Visual Examination, the 5 Row 1 tubes in Steam Generator #1 designated Columns 1, 2, 3, 4 and 5 which were abraded and 1 tube designated Row 1 Column 93 which was identified by Eddy Current Testing as having an indication in the tangent were plugged using Mechanical Plugs furnished by Westinghouse Corporation and installed by Westinghouse personnel using Westinghouse procedures approved for this activity.

On December 24, 1980 Unit #1 was removed from service for a brief outage scheduled primarily in order to perform an Ice Condenser Inlet Door Surveillance Inspection.

Based on cognizance of the problem experienced with the Unit #2 tube lane blocking devices, Plant Management and Senior American Electric Power Service Corporation Management made a decision to remove the tube lane blocking device hand hole covers and perform a visual inspection to determine the rigidity of the device and for evidence of tube abrasion. The decision was also rendered at this time not to employ an Eddy Current Testing technique to ascertain the percent of tube wall thinning but to plug tubes showing a significant external degradation.

The inspection revealed an appreciable amount of external damage to tubes in Steam Generator #1 designated Row 1 Columns 1, 3, 4, 5, 90, 91, 92 and 94 and in Steam Generator #4 Columns 1, 2, 3, 4, 5, 90, 91, 92, 93 and 94. Very light indications were noted in Steam Generator #2 on tubes designated Row 1 Columns 1, 2, 3, 4, 5, 90, 91, 93 and 94. The decision was made to plug tubes designated Row 1 Columns 1, 2, 3, 4, 5, 90, 91, 92, 93 and 94 in Steam Generators #1 and #4 using Mechanical Plugs furnished by Westinghouse Corporation installed by Westinghouse personnel in accordance with Westinghouse procedures approved for this activity.

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 pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59(a).

DC-12-1455

A pulsation dampner was added to the discharge piping of the Reciprocating Charging Pumps in the CVCS Letdown and Charging Systems of the Donald C. Cook Nuclear Plant. The purpose of this change was to reduce the excessive pump induced vibration of the piping downstream from the pump.

RFC DC-12-1455 is safety related since it introduces changes to a Class I System, namely, adding pulsation dampners to the piping downstream of the reciprocating charging pumps. Further, the change described herein is required to fulfill commitments made by us to the Nuclear Regulatory Commission in our letter dated August 8, 1978.

Based on the review of this RFC and the plans for performing the required analysis, it is concluded that this change does not pose an unreviewed safety question as discussed in 10 CFR Part 50.59c, nor will it compromise the health and safety of the general public.

DC-12-1767

Interim emergency sampling equipment was installed as required by NUREG-0578 to allow post-accident sampling of the Reactor Coolant System.

Additional switches for valves NRV-101 and 103 were installed in a sampling panel outside of the existing nuclear sampling room. The valve operators for valves NCR-105 and 106 were moved to the same sampling panel. Operation of valves NRV-101 and 103 can be accomplished at either location while NCR-105 and 106 can only be operated at the new sample panel outside the sampling room.

A sample line was installed after the failed fuel detector sample cooler and rerouted back to the CVCS Holdup Tanks. A sample is collected under a 4 valve configuration in a poly bottle housed in a shielded transport pig. All valve operation is via reach rod thru either an existing wall or lead shield wall. The sample will then be transported to the hot laboratory for analysis. A demineralized water flush line is provided to flush out the piping to reduce area radiation levels.

The installation of this change will enable the plant to perform reactor coolant sampling and analysis in the event the existing sampling room becomes uninhabitable.

This RFC was issued to meet a specific NRC requirement set forth in the recommendations of the short term lessons learned task force (NUREG-0578). While the system itself is not safety related, installation requires interfaces with safety grade structures. These interfaces include drilling into the auxiliary building wall and installing tubing onto the auxiliary building structure.

A safety review has been performed to determine the adequacy of the design and its ability to maintain radiation exposures within the limits of GDC19.

Based on this review, it is concluded that the change described herein, neither constitutes an unreviewed safety question as defined in 10 CFR 50.59 nor will it compromise the health and safety of the public.

<u>DC-12-1768</u>

Interim emergency sampling equipment was installed as required by NUREG-0578 to provide post-accident sampling of the containment atmosphere for Hydrogen, Radioiodines and Radioparticulates.

A sample line was installed to incorporate the normal R-11/12 inlet and outlet piping. This sample piping was routed to a 587' Auxiliary Building sampling location where a grab sample can be obtained in a shielded transport pig. A pump is provided in the sample piping configuration to return the sample flow to containment and insure a good sample. All sample valving and pump operation is by reach rod operated valves and a remote switch. Once collected, the gas sample is withdrawn via swagelock quick disconnects and transported to the hot laboratory for analysis.

The modification is required by NRC's short term lessons learned task force (NUREG-0578). However, the installation requires drilling into Class I structures (e.g. Auxiliary Building) and anchoring the lines to prevent the system from becoming a missile in the event of an earthquake. Because of these interfaces, this RFC is classified under the Safety Interface category.

A safety review has been performed to determine the adequacy of the design and its ability to maintain the exposure rates within the guidelines of GDC 19.

Based on this review, it is concluded that the change described herein, neither constitutes an unreviewed safety question as defined in 10 CFR 50.59 nor will it compromise the health and safety of the public.

DC-12-1771

Interim emergency sampling equipment was installed as required by NUREG-0578 to provide post-accident unit vent effluent monitoring.

A sample flow path is established by this change from the inlet to R31/32 thru a solenoid sampling valve, through a shielded cave and returned to the R31/32 exhaust via another solenoid valve by a pump. As the flow passes through the shielded cave, an access point allows radiation readings to be taken 6" from the line. A release rate can be calculated from these radiation levels.

This modification is required under the recommendations of NRC's short term lessons learned task force. There is no requirement for this system to be safety related. However, the installation requires activities such as core drilling of the auxiliary building wall which is a Seismic Class I structure. In addition these lines have to be anchored such that they will be held in place in the event of a design basis earthquake. These aspects bring the RFC under the Safety Interface category.

A review has been performed to determine the adequacy of the design and its ability to maintain associated exposure levels within the Guidelines of GDC 19.

Based on this review, it is concluded that the change described herein, neither constitutes an unreviewed safety question as defined in 10 CFR 50.59 nor will it compromise the health and safety of the public.

DC-12-2166 Unit #1 Only

Block and drain valves were added to the following systems on Unit #1 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. The installation of these valves was required to provide uniformity in the testing of the containment isolation valves in both Units.

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 10 CFR 50.59.

DC-12-2186

An additional Motor Driven Auxiliary Feedpump was installed in the existing Auxiliary Feedwater System in each unit of the Donald C. Cook Nuclear Plant. This modification separated the shared operation of the Auxiliary Feedwater System. In this manner, Units #1 and #2 will be able to meet the Limiting Condition of Operation for their respective Auxiliary Feedwater Systems, as prescribed by the Technical Specifications without reliance on equipment in the other Unit.

Prior to this installation, the loss of a Motor Driven Auxiliary Feedpump for more than 72 hours involved a Technical Specification Action Statement which required the shutdown of both Units.

This modification is further described in Section 10 of the FSAR.

This RFC is considered to be safety related because the Auxiliary Feedwater System is part of Engineered Safety Features (ESF), is Seismic Class I and is required to function during design basis accidents.

The Auxiliary Feedwater System provides the necessary heat sink for the Reactor Coolant System (RCS) through the steam generators, under transient and accident conditions. The analysis of design basis conditions demonstrate the critical role of the Auxiliary Feedwater System. As such, the design must provide redundancy and diversity. At the same time, the Auxiliary Feedwater System is also designed to provide feedwater for normal plant startup and cooldown.

To eliminate the impact on plant operation of the conflictive aspects of Cook's Technical Specifications, this RFC proposes permanent modifications. The modified Auxiliary Feedwater System would only require one Unit to be shutdown to meet surveillance requirements.

The Nuclear Safety & Licensing Section has reviewed the subject RFC in light of current applicable Regulatory Guides, Standard Review Plans, and Branch Technical Positions including fire protection and security requirements. The results of this review are summarized below:

1. System components and piping will have sufficient physical separation and/or shielding to protect the essential portions of the system from the effects of internally and externally generated missiles.

2. The system design accounts for the effects of pipe whip and jet impingement that may result from high or moderate energy breaks or cracks.

3. The system and components satisfy design code requirements, as appropriate for the assigned quality group and seismic classifications.

4. The failure of non-essential equipment or components does not affect essential functions of the system.

5. The system is capable of withstanding a single active failure.

6. The system posesses diversity in motive power sources such that system performance requirements may be met with any of the assigned power sources.

7. The system design precludes the occurrence of fluid flow instabilities, e.g., water hammer, in system inlet piping during normal plant operation or during upset or accident conditions.

8. Functional capability is assured by suitable protection during abnormally high lake water levels such as the maximum probable seiche.

9. The capability exists to detect and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.

10. Provisions are made for operational testing.

11. Instrumentation and control features are provided to verify the system is operating in a correct mode.

12. The technical specifications are such as to assure the continued reliability of the Auxiliary Feedwater System during plant operation.

Therefore, the Nuclear Safety and Licensing Section has no reason to object to the modifications being proposed under the subject RFC.

The required FSAR revisions and Technical Specification changes have been submitted to the NRC.

RFC DC-12-2186 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10 CFR 50.59. This change will not, in any way, adversely effect the health and safety of the public.

DC-12-2196

Two additional Gas Decay Tanks were installed to supplement the six existing tanks in the Waste Gas System. The piping, valves, instrumentation and controls installed for the new tanks is identical in design to the existing tanks.

These additional tanks were required to allow more flexibility in the operation of the Waste Gas System during refueling or shutdown of either Unit #1 or Unit #2.

RFC DC-12-2196 was reviewed and found to be acceptable. Provisions were made for these two gas decay tanks in the original design in the terms of electrical connections and support pedestals and thus only the tanks are being installed now. These tanks are being designed and installed to the original qualification criteria, i.e. Seismic Class I.

It is concluded that this addition of two tanks or the design of the tanks themselves do not constitute an unreviewed safety question as defined in 10 CFR 50.59.

DC-12-2222 Unit #1 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 a "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 has 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 10 CFR 50.59. This change will not adversely affect the health and safety of the public.

DC-12-2225 Unit #1 Only

A Reactor Coolant Pump Motor oil spillage protection and control system was installed on all 4 reactor coolant pump motors of Unit #1.

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 unreviewed safety question as defined in the 10 CFR 50.59. This RFC further enhances the fire protection systems in the Cook Nuclear Plant.

DC-12-2276 Unit #1 Only

Local Control capabilities were provided for the Unit #1 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 10 CFR 50.59.

DC-01-2313

Backup molded case circuit breakers and/or fuses were added for 600V ESS and Non-ESS electrical equipment inside the Unit #1 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-01-2313 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.

Since the installation of this redundant equipment will increase the protection to the containment electrical penetrations during electrical faults, it does not constitute an unreviewed safety question as defined in 10 CFR 50.59.

DC-12-2315

The capacity of the Spent Fuel Pit at the Donald C. Cook Nuclear Plant was increased from the initial design of 500 fuel assemblies to 2050 assemblies. This was accomplished by replacing the existing spent fuel racks with poisoned high density racks.

The replacement spent fuel storage racks were fabricated primarily from type 304 stainless steel. The individual fuel assemblies will be stored in square fuel storage cells fabricated from stainless steel-clad Boral material. The high density (poison) spent fuel module construction is essentially a replica of the design used in the replacement racks for the Salem Nuclear Generating Station.

The design utilizes a stiffened module base and an upper box structure consisting of plate diaphragms and a top grid. The vertical loads are carried by the module base. Horizontal seismic loads are carried to the module base through the plate diaphragms. Tipping is prevented by coupling adjacent racks through a bolted connection at the top grid level.

The detailed design of the spent fuel storage cells is slightly different from the design for the Salem Nuclear Generating Station. Their basic function and construction, however, are similar. Each cell is a square cross-section formed from an inner shroud of stainless steel, a center sheet of aluminum clad B_4C, and an outer shroud of stainless steel. This cell acts as a storage space and, in addition, provides sufficient neutron absorption by the boron carbide contained in the Boral sheet to allow spacing of spent fuel in a 10.5 inch by 10.5 inch array. The fuel weight is carried directly on the module base. A flared guide and transition section is provided at the top of each storage cell. This transition is designed to assure ease of entry and to preclude fuel assembly hang-up and damage.

These replacement spent fuel storage racks will provide storage capacity and allow for the continued operation of both Unit No. 1 and Unit No. 2 until approximately the first part of 1992 while still maintaining the capacity for a full core discharge reserve (FCDR) of 193 locations.

Correspondence between AEPSC and the NRC (AEP:NRC 00105, 00116, 00169, 00213B, C, D) provides additional information pertaining to the Spent Fuel Rack replacement.

Interim approval of the installation of Greater Capacity fuel racks under RFC Charge No. DC-12-2315 by the Nuclear Safety and Licensing Section was given previously, provided certain concerns were addressed and satisfactorily met prior to installation of the Fuel Racks. Analyses of those items of concern with respect to their compatability to the present features of our facility have been performed. Documentation substantiating that these analyses satisfactorily meet the safety criteria and concerns with respect to compatibility with the present features of our facility is listed below:

1. Criticality - AEP:NRC Letter No. 00105, Section 3.1
   Dated November 22, 1978

2. Thermal - AEP:NRC Letter No. 00116 Section 3.5
   Dated January 22, 1979

3. Seismic - AEP:NRC Letter No. 00169 Section 3.6
   Dated April 16, 1979

4. Accidents - AEP:NRC Letter No. 00169 Sections 3.6.5 & 3.6.6 Dated April 16, 1979

5. Movement of Heavy Loads over the Spent Fuel Pool - Technical Specification No. 3.9.7 Amendments to the Licenses No.'s DPR-58-32 for Unit 1 dated October 16, 1979 and DPR-74-13 for Unit 2 dated also on October 16, 1979.

6. NRC Review & Approval to ensure that the proposed modification does not now constitute an unreviewed safety question as defined in 10 CFR Section 50.59.
NRC Safety Evaluation Report dated October 16, 1979 and License Amendments No.'s DPR-58-32 for Unit 1 and DPR-74-13 for Unit 2.

Hence, this RFC is now accepted by the Nuclear Safety and Licensing Section.

<u>DC-12-2435 (Unit #1 Only)</u>

Thermal sleeves were installed at the junction of the 16" diameter elbows and the Steam Generator nozzles in the Feedwater System on Unit #1 of the Donald C. Cook Nuclear Plant. The thermal sleeves were installed to minimize thermal stresses in the elbows.

The feedwater line thermal sleeve modification extends from the vertical pipe reducer through the elbow and into the steam generator nozzle. The nozzle end of the thermal sleeve contains two piston rings (contained in one groove) to seal the annular gap. This promotes a low convective heat transfer coefficient which is beneficial in reducing thermal stresses at the nozzle and pipe inside surfaces.

The nozzle thermal sleeve is 0.38" thick and fabricated from SA-106-GR B. The new thermal sleeve is 0.50" thick and also fabricated from SA-106-GR B carbon steel. An inconel 600 weld build-up is placed on the pipe reducer to accomodate welding of the new sleeve. This feature provides an improvement in fatigue strength over that of an equivalent carbon steel section.

This change is considered to be safety-related due to the fact that the affected portion of the main feedwater piping, the steam generators, and the thermal sleeves themselves are Seismic Category I components.

Nuclear Safety and Licensing has continuously been involved in the efforts regarding the feedwater elbow cracking problem, the subsequent correspondence between AEP and the NRC, and with regards to the corrective actions taken. Based on the information contained in Attachment No. 2 to

our AEP:NRC:00305 submittal and completion of an acceptable seismic analysis, it has been concluded that implementation of this RFC does not constitute an unreviewed safety question as defined in 10 CFR 50.59 and will, in fact, increase the already high level of safety of the Cook Plant.

DC-12-2441

Dedicated saturation meters were installed in both Units of the Donald C. Cook Nuclear Plant. The saturation meter is a Babcock and Wilcox device which is connected to two Reactor Coolant System wide range pressure transmitters, eight plant incore thermocouples and eight plant-wide range RTD's (4 hotleg and 4 coldleg). A remote display meter located in the Control Room on the Boric Acid Panel will display margin to saturation temperature (Tsat) or margin to saturation pressure (Psat). An annunciator will be actuated if the alarm setpoint is reached.

The meter provides an on-line aid to the operator to help him assure that an adequate "saturation margin" is maintained in the plant at all times.

This modification is being implemented in accordance with the requirements of NUREG-0578, Item 2.1.3.b, as amended by NUREG-0737. This RFC is considered safety-related because the system is required for the detection of inadequate core cooling. In addition, the RCS pressure and temperature input signals are generated by Class IE electrical components. The system design and installation are in conformance with the NRC requirements in this area. This RFC does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10 CFR 50.59.

DC-12-2476

The 4KV Safety (ESS) buses undervoltage protection scheme was modified to ensure separation of the safety buses from the offsite power source during degraged grid voltage conditions as postulated in the NRC communication on "Adequacy of Station Electric Distribution System Voltages."

Details of this change are as follows:

1. Remove "Degraded Grid Voltage" tripping of feed breakers to 4KV ESS buses. Degraded grid voltage relays mounted on the control room auxiliary panels were moved to the 4KV room. Only alarm "34.5 KV Tr. Voltage Low" and white lights on the auxiliary panels are operated when these relays now operate. Set-point is 90% nominal.

2. Added "Degraded Bus Voltage" tripping of feed breakers to 4KV ESS buses. The "degraded bus voltage" relays which are in the 4KV room and only provided an alarm were moved to the control room auxiliary panels and will now trip the feed breaker to the 4KV ESS buses when 2 out of 3 phases operate. These relays were reset to operate at 80% nominal bus voltage. Pump buses "T_ _A" and T_ _D are monitored. To allow for expected bus voltage drops when starting large motors, a 2 minute time delay was added before tripping open the feed breakers to all 4KV ESS buses. Tripping function of the "degraded bus voltage" relays is disabled when not on normal reserve feed.

3. Increased Diesel Generator and load shedding relay actuation from 60% nominal to 80% nominal. To provide added instant protection if 4KV ESS bus voltage should drop below 80% nominal, Diesel Generator start and load shedding relay's setpoints were increased to 80% nominal. To eliminate possible switching transients or 600 volt faults from initiating DG start and load shedding, a 2 second time delay was added.

In summary, if 4KV ESS bus voltage on any pump bus should drop below 90% while on reserve feed in excess of 2 minutes, the feed breakers to all 4KV ESS buses will open. At any time of any 4KV ESS bus voltage drops to 80% in excess of 2 seconds, DG startup and load shedding will occur.

The subject RFC calls for the revision of the 4KV safety bus undervoltage protection logic. The 4KV safety buses are Seismic Category I components and their undervoltage and 'loss of power' setpoints technical specification items. Thus, this RFC is considered "safety-related."

The revised setpoints and undervoltage protection scheme called for in this RFC are based on an analysis performed by members of the Electric Engineering Division. The results of this analysis were forwarded to the NRC via our AEP:NRC:00268 submittal dated December 17, 1979.

The "grid degraded voltage" and "loss of power" setpoints for the 4KV safety buses have been chosen so as to assure that the voltage at the 4KV safety buses is sufficient to power the applicable safety-related equipment during abnormal grid voltage conditions while minimizing the potential challenges to the emergency diesel generators; thereby maximizing the use of the available offsite power source.

Implementation of this RFC does not constitute an unreviewed safety question as defined in 10 CFR 50.59.