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AUTH.NAME AUTHOR AFFILIATION
HUNTER,R.S. Indiana & Michigan Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards response to NRC 801031 ltr re implementation of post-TMI requirements contained in NUREG-0737.

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INDIANA & MICHIGAN ELECTRIC COMPANY
P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

January 8, 1981
AEP:NRC:0398

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
Post-TMI Requirements (NUREG-0737)

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

The attachments to this letter provide our response to Mr. D. G. Eisenhut's letter of October 31, 1980, which we received on November 6, 1980, concerning the implementation of the post-TMI requirements contained in NUREG-0737. Our responses follow the same format of Enclosure 1 of NUREG-0737. For those items of Enclosure 1 which are implemented under NUREG-0578 and are noted as "COMPLETE", no response is provided.

We have proceeded with the implementation of many of the post-TMI requirements based on the criteria of NUREG-0578. In some instances a proposed revision to Regulatory Guide 1.97 was referenced for the design and qualification criteria of certain equipment. However, NUREG-0737 now requires us to meet the provisions of its Appendix B for design and qualification of some of the same equipment. We are told in the applicable items that this is to be considered a new requirement. We do not believe that the issuance of new requirements at this late date is compatible a priori with your implementation schedules and with our established engineering practices. Appendix B of NUREG-0737 represents a substantial change in the design and qualification of some equipment to the extent that it may impact on the completion of items already underway. Where this is the case, it is pointed out in the attachments to this letter. Even more, our assessment of the impact of the design and qualification criteria of Appendix B is still under review. Any additional problem areas will be noted in supplemental correspondence to this letter.

Very truly yours,

R. S. Hunter
Vice President

A001 s/1

cc: attached

8101130217

cc: R. C. Callen
G. Charnoff
John E. Dolan
R. W. Jurgensen
D. V. Shaller - Bridgman
NRC Region III Resident Inspector at Cook Plant - Bridgman

STATE OF NEW YORK

COUNTY OF NEW YORK

R. S. Hunter, being duly sworn, deposes and says that he is the Vice President of Licensee Indiana & Michigan Electric Company, that he has read the foregoing response to the post-TMI requirements contained in NUREG-0737 and recognizes the contents thereof; and that said contents are true to the best of his knowledge and belief.

[signature]

Subscribed and sworn to before me this 9th day of January, 1981.

Kathleen Barry

[illegible]
NOTARY PUBLIC, State of New York
No. [illegible]
Qualified in Queens County
Certificate filed in New York County
Commission Expires March 30, 1981

ATTACHMENT NO. 1

TO

AEP:NRC:00398

RESPONSE TO ITEM I.A.1.1, SHIFT TECHNICAL ADVISOR;

The Shift Technical Advisor Training Program as described below is complete with dedicated personnel on shift as of January 1, 1981.

The Shift Technical Advisor (STA) Training Program was developed to provide the necessary training and background for both the accident assessment function and an operating experience assessment function. The initial training program for these functions consisted of the following subjects in addition to extensive systems training;

- Reactor Physics, Chemistry and Materials
- Reactor Thermodynamics, Fluid Mechanics, and Heat Transfer
- Electrical Engineering including Reactor Control Theory

All subjects were taught at the college level by a local university, NRC approved vendors, and the on-site Training Department staff.

Only personnel with prior degrees in engineering or physical sciences are selected and, as such, their prior training in mathematics was determined sufficient to meet the requirements of Mr. Denton's October 30, 1979 letter. To ensure the STAs familiarity with design, function, arrangement, and operation of plant systems as required by the October 30, 1979 letter, only applicants with prior power plant experience will be admitted into the STA program. In addition, the extensive system training given in plant systems, including design and operation is consistent with the INPO recommendations for systems training.

Transient and accident response training will be given through special lectures by the NSSS vendor and simulator training. This training is consistent with the requirements of the October 30, 1979 letter.

Attachment I.A-1 shows a comparison of the content of the established D. C. Cook Plant STA Training Program to the INPO guidelines contained in Appendix C of NUREG-0737. All major areas of concern have been covered. In some areas, the INPO recommended contact hours (Section 6 of Appendix C) cannot be met due to time and resource constraints. However, we feel that the content of all areas of concern are adequately covered to meet the training qualifications as specified in the October 30, 1979 NRC letter.

The STA candidates will attend a simulator training program annually where they will participate in plant evolutions to gain experience in situation assessment and the necessary actions to mitigate the consequences of an accident.

An STA Requalification Program is designed to maintain a continuous degree of knowledge and proficiency as required by ANS 3.1 is established for the Donald C. Cook Nuclear Plant Units 1 and 2.
This program shall apply to all STA's, including STA's who perform such duties on an infrequent basis. A site appointed Training Coordinator has been assigned to implement and administer this program. A brief description of the requalification program is contained in Attachment I.A-2.

RESPONSE TO ITEM I.A.1.3, SHIFT MANNING:

1. Limit Overtime -

Plant Manager Standing Order PMSO.054 dated October 29, 1980 was issued limiting overtime for Reactor Operators, Senior Reactor Operators and Shift Technical Supervisors.

2. Shift Staffing -

Implemented as described in our November 7, 1980 letter (AEP:NRC:00450) in response to Eisenhut's letter of July 31, 1980 concerning interim criteria for shift staffing.

RESPONSE TO ITEM I.A.3.1, QUALIFICATIONS OF REACTOR OPERATORS:

Implemented as described in our September 11, 1980 letter (AEP:NRC:00395) in response to Denton's letter of March 28, 1980 concerning reactor operator qualifications program.

RESPONSE TO ITEM I.C.1, SHORT-TERM ACCIDENT AND PROCEDURES REVIEW:

The Westinghouse Owners Group (WOG) of which Indiana & Michigan Electric Company (I&MECo) is a member will submit by January 1, 1981, a detailed description of the program to comply with the requirements of Item I.C.1.

In addition, we have revised our natural circulation cooldown procedure to limit the cooldown rate to prevent the formation of bubbles in the reactor head area. This revision is a result of our review of IE Circular No. 80-15.

RESPONSE TO ITEM I.C.5, FEEDBACK OF OPERATING EXPERIENCE:

In accordance with our letter of June 20, 1980 (AEP:NRC:00419), which respondedto Eisenhut's May 7, 1980 letter concerning additional TMI-2 requirements, procedures for feedback of operating experience to plant staff have been reviewed. Existing, in place plant procedures were found to adequately address all requirements for NRC reports and notices and Cook Plant Condition Reports as stated in Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff".

A formal new procedure which meets the requirements stated in Item I.C.5 has been implemented at the Plant for the handling of non-NRC Notices, such as NSSS Vendor Notices, and INPO Notices.

Consistent with the requirements for an 'Operating Experience Assessment' group as outlined in Mr. Denton's October 30, 1979 letter to all operating nuclear plants, procedures for handling NRC, vendor and industry-related notices will be modified to directly involve the 'STA in office duty' in an immediate safety assessment function. If it is found that the notice contains information of significant importance that should not wait for emphasis through the usual routing, this group will act promptly to take appropriate action. All NRC, vendors, industry and internal Cook Plant Notices will be reviewed by the 'STA in office duty' for trend analysis.

RESPONSE TO ITEM I.C.6, VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES:

This requirement, formally issued by NUREG-0737, has been reviewed for its impact on Cook Plant operating activities. We currently have in place an effective system of verifying the correct performance of operating activities, the essential elements of which are described below.

Conducting surveillance on the Reactor Protection and Engineered Safeguards System during plant operation requires Shift Supervisor or Operating Engineer approval prior to start.

Plant surveillance testing is in accordance with approved schedules and procedures. Data are reviewed by the Unit Supervisor, Operating Engineer or Shift Operating Engineer. The daily master schedule for completion of required surveillance is reviewed by the Operating Engineer or Shift Operating Engineer to determine current status.

Clearance Permits to remove essential equipment from service are prepared by an Operating Engineer or by the Shift Operating Engineers. To insure that the ControlRoom is aware of equipment status, the permits are sent to the Unit Supervisor who directs their implementation. Return to service is in the reverse order of the above with the final acceptance for system operation being made by the Operating Engineer or Shift Operating Engineer. System lineup of the safety-related system requires independent verification of valve lineup after it has been out of service for maintenance or following surveillance testing.

RESPONSE TO ITEM I.D.1, CONTROL ROOM DESIGN REVIEW:

An assessment of the requirements for implementation of this item will be performed when NUREG-0700 is issued and reviewed.

RESPONSE TO ITEM I.D.2, PLANT SAFETY PARAMETER DISPLAY CONSOLE:

We are in the process of implementing the Westinghouse designed Plant Safety Status Display (PSSD) system in the control rooms of the Cook Nuclear Plant. On May 27, 1980, AEPSC representatives met with Messrs. Hanauer, Mattson and other NRC staff members and presented to them a description of the PSSD to be installed in Cook Plant. This equipment has been ordered since May 9, 1980 and we are proceeding ahead with the work required to complete its' installation and operation. The PSSD is further described in Westinghouse's submittal of WCAP-9725 to the NRC dated June 13, 1980 (NS-TMA-2261). This system is an integral part of the Westinghouse Technical Support Center Complex and is designed with the application of Human Engineering principles.

RESPONSE TO ITEM II.B.1, REACTOR COOLANT SYSTEM VENTS:

The documentation required for this item will be provided by July 1, 1981. Installation of this system will be completed by July 1, 1982. This schedule complies with the requirements of NUREG-0737 for this item.

RESPONSE TO ITEM II.B.2, DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT:

This item has been implemented by the responses provided in our letters dated March 10, 1980 (AEP:NRC:00334B) and May 15, 1980 (AEP:NRC:00334D) regarding item 2.1.6.b of NUREG-0578.

In addition, the specific equipment qualifications are provided in our submittals in response to IE Bulletin 79-01B, dated March 7, 1980 (AEP:NRC:00356), May 7, 1980 (AEP:NRC:00356A), June 5, 1980 (AEP:NRC:00356B) and October 31, 1980 (AEP:NRC:00356C).

RESPONSE TO ITEM II.B.3, POST ACCIDENT SAMPLING:

Design modifications, sampling equipment and analytical capability will be in place to satisfy this requirement by January 1, 1982, the NUREG-0737 given date.

RESPONSE TO ITEM II.B.4, TRAINING FOR MITIGATING CORE DAMAGE:

We plan to use the Westinghouse program entitled "Mitigating Core Damage Training Course". This program will be initiated around April 1, 1981 and completed by October 1, 1981 in accordance with the requirements of this item.

RESPONSE TO ITEM II.D.1, PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES (NUREG-0578, Item 2.1.2):

As a participating member of the EPRI PWR Safety and Relief Valve Test Program, Indiana & Michigan Electric Company is complying with the requirements of NUREG-0578, Item 2.1.2. By letter dated December 15, 1980, R. C. Youngdahl of Consumers Power Company has provided the current participating utility positions with regard to the clarifications of item II.D.1 of NUREG-0737.

RESPONSE TO ITEM II.D.3, DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION:

This item has been implemented in the Cook Plant and our compliance is documented in the NRC Safety Evaluation Report issued on March 20, 1980. The human factors analysis recommended in clarification item 6 will be included in the detailed control room design review to be performed in accordance with our response to item I.D.1 of NUREG-0737.

RESPONSE TO ITEM II.E.1.1, AUXILIARY FEEDWATER SYSTEM EVALUATION:

The information required by this item concerning the AFW system flow design basis has been submitted by our letter dated November 3, 1980 (AEP:NRC:00300C). The NRC has closed this item with the issuance of the October 6, 1980 Safety Evaluation Report of the Cook Plant's AFW system reliability.

RESPONSE TO ITEM II.E.1.2, AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION:

Part 1: The AFW system is automatically started at the Cook Plant and the initiation signals and circuits meet safety grade requirements.

Part 2: The flow indication system to be used at D. C. Cook Plant will be one auxiliary feedwater flow rate indicator and one narrow range steam generator level indicator for each steam generator. Installation of environmentally qualified auxiliary feedwater flow rate transmitters will be compatible with the NUREG-0737 implementation schedule.

The information previously submitted for part 1 of this item and our letter of December 10, 1980 (AEP:NRC:00307E) in response to S. A. Varga's letter of October 31, 1980 shows compliance with the specified requirements of part 2 of this item. Also, our submittal of the Category A Lessons Learned Technical Specifications by letter dated December 10, 1980 (AEP:NRC:00449) is consistent with the above implementation of AFW flow indication.

RESPONSE TO ITEM II.E.4.2, CONTAINMENT ISOLATION DEPENDABILITY:

D. C. Cook Plant is already in compliance with NRC positions 1 through 4 as documented in the NRC March 20, 1980 SER.

The Cook Plant design basis for the minimum pressure setpoint of containment isolation is in compliance with NRC position 5. For D. C. Cook Plant each unit has two levels of containment isolation identified as Phase A and Phase B. Phase A isolation closes all lines penetrating the containment except essential lines such as Safety Injection and Containment Spray which are not isolated, and component cooling water to the reactor pumps and service water to the ventilation units which isolates on Phase B. Phase A isolation is initiated by containment pressure high (1.1 psig), any safety injection signal and manually. The D. C. Cook Plant Technical Specification for the high limit of containment internal pressure has as its Limiting Condition of Operation (LCO), +0.3 psig. Therefore, the differential between the LCO and the Phase A isolation setpoints from containment pressure high (0.8 psig) is within the requirements of clarification item (6) for a margin of 1 psi.

Thus, the 1.1 psig setpoint is the minimum compatible with normal operating conditions and no further action is required.

The requirements contained in NRC positions 6 and 7 have already been implemented as the Cook Plant. These two positions have been addressed as indicated in Attachment II.E-1.

RESPONSE TO ITEM II.F.1, ATTACHMENT 1, NOBLE GAS MONITOR:

The upgraded radiation monitoring system at D. C. Cook Plant is being designed to provide the information required by both Attachments 1 and 3 of Item II.F.1. This system which is on order and being manufactured has been purchased in accordance with your earlier requirements of NUREG-0578, prior to issuance of NUREG-0737, based on the commitment in our October 24, 1979 letter (AEP:NRC:00253). The design of the system to meet the criteria specified in Attachments 1 and 3 is interrelated to the extent that separation of the documentation would be extremely difficult. Schedule relief of the January 1, 1981 date is therefore requested to provide the documentation for Attachment 1 on the same schedule as that required by Attachment 3, i.e. by July 1, 1981 rather than January 1, 1981. The portions of our upgraded radiation monitoring system for compliance with this item will be installed by January 1, 1982 consistent with the requirements of Attachment 1.

The upgraded radiation monitoring system at Cook Plant will meet the requirements of the NRC positions of Attachment 1 when reviewed in total but not on an individual monitor bases. The system will provide the ranges for the release path as stipulated by Table II.F.1-1 under "Design Basis Minimum Range". However, the system normal operating range monitors are not capable of this extended range nor are the post-accident monitors capable of the ALARA range requirements. We wish to point out that clarification 4 regarding ALARA range requirements on the post-accident monitors conflicts with the requirements of clarification 1 and Table II.F.1-1.

RESPONSE TO ITEM II.F.1, ATTACHMENT 2, SAMPLING AND ANALYSIS OF PLANT EFFLUENTS:

The upgraded radiation monitoring system at D. C. Cook Plant is being designed to provide information as required by Attachment 2. The proposed Hi range noble gas monitors will be able to sample particulates and radioiodines by absorpiton on a filter, followed by an onsite laboratory analysis.

The absorption of radio-iodines will be done on a Silver Zeolite filter. We believe that the technology currently available is limited to low temperature gases with low moisture content. Consequently, steam vents are not monitored for Iodine and particulates at the Cook Plant. The D. C. Cook Plant will be equipped with facilities to analyze these filters consistent with clarifications 1 and 2.

The equipment for implementation of this requirement is on order and is scheduled for delivery in late 1981. Therefore, the first available date for installation is the refueling outages in 1982. As such, an extension is requested beyond the implementation date of January 1, 1982 until these refueling outages.

RESPONSE TO ITEM II.F.1, ATTACHMENT 3, CONTAINMENT HIGH RANGE RADIATION MONITOR:

As stated in the above response to Item II.F.1, Attachment 1, the radiation monitoring system upgrade for D. C. Cook Plant is being designed to provide the required information of both this attachment and Attachment 1. However, since Attachment 3 stipulates significant changes from previous documents, a detailed review of the impact of these changes cannot be provided in this submittal. Specifically, the impact of Appendix B to NUREG-0737 regarding Design and Qualification of Equipment is still under review. If deviation of this Attachment's positions or clarifications are necessary, a detailed explanation of and justification for the deviation will be provided by July 1, 1981, along with the documentation required for final review. This system which is on order and being manufactured has been purchased in accordance with your earlier requirements of NUREG-0578, prior to issuance of NUREG-0737. The portions of our upgraded radiation monitoring system for compliance with this item will be installed by January 1, 1982 consistent with the requirements of Attachment 3.

RESPONSE TO ITEM II.F.1, ATTACHMENT 4, CONTAINMENT PRESSURE MONITOR:

This item has been implemented as per our letters of October 24, 1979 (AEP:NRC:00253), and January 18, 1980 (AEP:NRC:00334).

RESPONSE TO ITEM II.F.1, ATTACHMENT 5, CONTAINMENT WATER LEVEL:

The containment water level monitoring system which was purchased for Cook Plant will be implemented in accordance with our letters of October 24, 1979 (AEP:NRC:00253) and January 18, 1980 (AEP:NRC:00334). The required installation and documentation will be completed by January 1, 1982.

RESPONSE TO ITEM II.F.1, ATTACHMENT 6, CONTAINMENT HYDROGEN MONITOR:

Design modifications and monitoring equipment that will satisfy this requirement will be installed and operational by January 1, 1982. The required documentation will be submitted by January 1, 1982.

RESPONSE TO ITEM II.F.2, INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING:

As stated in our previous submittals addressing Item 2.1.3.b of NUREG-0578, letters dated January 18, 1980 (AEP:NRC:00334), Attachment 6 and March 10, 1980 (AEP:NRC:00334B), D. C. Cook Plant is implementing an inadequate core cooling (ICC) system. The system is to be composed of reactor water level indication supplied by Westinghouse Electric Corp. supplemented as necessary by the equipment available to monitor margin to saturation of the reactor coolant system. The position, clarification, attachments and appendices now imposed by this item represent a major significant change to those previously required by other documents. Westinghouse Electric Corp. is presently preparing Topical Reports to respond to the documentation requirements of this item. A tentative schedule for submittal of these reports by Westinghouse to the NRC is January 1, 1981. A detailed review by AEPSC cannot be performed by the January 1, 1981 documentation deadline for this item. Relief to April 1, 1981 is therefore requested from the documentation required date of January 1, 1981 to allow for detailed AEPSC review of these Westinghouse reports. The reactor water level system will be installed consistent with the implementation schedule of this item.

Our existing core exit thermocouple system does not meet the requirements of Attachment 1 to Item II.F.2.

RESPONSE TO ITEM II.K.2.13, THERMAL MECHANICAL REPORT --- EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER:

A program will be completed and documented to the NRC by January 1, 1982 by the Westinghouse Owners Group to completely address the NRC requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. This program will consist of analysis for generic W PWR Plant groupings.

RESPONSE TO ITEM II.K.2.17, POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEMS DURING TRANSIENTS:

The WOG is currently addressing the potential for void formation in the Reactor Coolant System (RCS) during natural circulation cooldown conditions, as described in Westinghouse letter NS-TMA-2298 (T. M. Anderson, W. to P.S. Check, NRC). A report describing the results of this effort will be provided to the NRC before January 1, 1982.

RESPONSE TO ITEM II.K.2.19, SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS:

The transient analysis code, LOFTRAN, and the present small break evaluations analysis code, WFLASH, have both undergone benchmarking against plant information or experimental test facilities. These codes under appropriate conditions have also been compared with each other. The WOG will provide on a schedule consistent with requirement of item II.K.2.19 a report addressing the benchmarking of these codes.

RESPONSE TO ITEMS II.K.3.1 - INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM; AND II.K.3.2 - REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM:

The WOG is in the process of developing a report to address the NRC

concerns of Item II.K.3.2. However, due to the time-consuming process of data gathering, breakdown, and evaluation, this report is scheduled for submittal to the NRC on March 1, 1981. As required by the NRC, this report will be used to support a decision on the necessity of incorporating an automatic PORV isolation system as specified in Item II.K.3.1.

RESPONSE TO ITEM II.K.3.5, AUTOMATIC TRIP OF REACTOR COOLANT PUMP DURING LOSS OF COOLANT ACCIDENT:

The WOG resolution of this issue has been to perform analyses using the Westinghouse small break evaluation model (WFLASH) to show that ample time is available for the operator to trip the reactor coolant pumps following certain size small breaks, see WCAP-9584. This is in accordance with our letter of June 20, 1980 (AEP:NRC:00419) which responded to Mr. Eisenhut's May 7, 1980 letter concerning additional TMI-2 requirements.
In addition the Owners Group is supporting a best estimate study using the NOTRUMP computer code to demonstrate that tripping the reactor coolant pump at the worst trip time, after a small break will lead to acceptable results.

For both of these analysis efforts, the WOG is performing blind post-test predictions of loft experiment L3-6. The input data and model to be used with WFLASH on test L3-6 has been submitted to the staff on 12/1/80 (NS-TMA-2348), the information to be used with NOTRUMP on test L3-6 will be submitted prior to performance of the test as stated in letter OG-45 dated 12/3/80.

The Loft prediction from both models will be submitted to the staff on February 15, 1981 given that the test is performed on schedule. The best estimate study is scheduled for completion by April 1, 1981.

Based on these studies, the WOG believes that resolution of this issue will be achieved without any design modifications. In the event that this is not the case, a schedule will be provided for potential modifications.

RESPONSE TO ITEMS II.K.3.9, PID CONTROLLER; II.K.3.10, PROPOSED ANTICIPATORY TRIP MODIFICATIONS; II.K.3.12, ANTICIPATORY TRIP ON TURBINE TRIP:

These items have been implemented per our June 20, 1980 letter (AEP:NRC:00419).

RESPONSE TO ITEM II.K.3.17, EMERGENCY CORE COOLING SYSTEM OUTAGES:

The following information provides our report on ECC system outages committed to in our letter of June 20, 1980 (AEP:NRC:00419) in response to this item. Shown in the attached tables are outage times for Centrifugal Charging, Residual Heat Removal, Safety Injection, Containment Spray and the Emergency

Diesel Generator Systems while the reactor was in Modes 1, 2, 3 or 4. Separate listings are given for Units 1 and 2 starting from initial criticality, 1/18/75 and 3/10/78 respectively, as shown in Tables II.K.3-1 and II.K.3-2 which are attached.

In some instances the exact length of time a component was out of service could not be determined. For these cases it was assumed that it was out half of the Technical Specifications maximum allowable time. These cases are marked with an asterisk.

We believe that Technical Specifications for the ECC systems in Cook Plant are sufficiently restrictive to keep control of their availability. No further actions are required for this item.

RESPONSE TO ITEM II.K.3.25, EFFECT OF LOSS OF AC POWER ON RCP SEALS:

The D. C. Cook Plant is not susceptible to seal failure upon loss of off-site power. The loss of off-site power results in a trip of the Reactor Coolant Pumps, but the Centrifugal Charging Pumps (CCP's) continue to supply seal injection to the RCP's upon loss of off-site power. The CCP's are powered from the Emergency Diesel Generators.

RESPONSE TO ITEMS II.K.3.30, SMALL BREAK LOCA METHODS AND II.K.3.31, COMPLIANCE WITH 10 CFR 50.46:

These items have been implemented as per our letters of June 20, 1980 (AEP:NRC:00419) and October 1, 1980 (AEP:NRC:00398A).

RESPONSE TO ITEM III.A.1.2, UPGRADE EMERGENCY RESPONSE FACILITIES:

I&MECo is presently reviewing its emergency support centers versus the requirements of NUREG-0654 and the results of the review will be part of a revised and upgraded Facility Emergency Plan (EP) for the Donald C. Cook Nuclear Plant. This plan will be submitted to the NRC and will address the upgraded emergency planning rules of 10 CFR 50.54 and Appendix E to 10 CFR 50 (item III.A.2 below). We have submitted significant comments to the Commission on the draft of NUREG-0696 and we await issuance of the final criteria.

RESPONSE TO ITEM III.A.2, IMPROVING LICENSEE EMERGENCY PREPAREDNESS - LONG TERM:

Revision 1 to NUREG-0654 was published in November, 1980. We are in receipt of this revision and are in the process of reviewing and revising our EP, where necessary, to address the changed criteria indicated in Revision 1 to NUREG-0654. Due to the time required for management review, final typing and printing we have sent the NRC a separate letter requesting an extension of the submittal date for this report to January 26, 1981. See our letter of December 30, 1980 (AEP:NRC:0308C).

RESPONSE TO ITEM III.D.3.3, IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS:

The following cart mounted, continuous air monitors are available for use in an emergency that may involve airborne radioactivity concerns:

Two Nuclear Measurements Corporation, Model AM-221, which monitors air-borne particulate (beta scintillator detector) and radioiodine (Silver Zeolite cartridge and single channel analyzer calibrated to 365 KeV). These units are dedicated to emergency use in the Technical Support Center and either Control Room.

Two Eberline Model PING-1 airborne particulate and radioiodine monitor. Radioiodine monitor, usually used with TEDA impregnated charcoal, will also accept the Silver Zeolite cartridge. The detector is connected to single channel analyzer calibrated to 365 KeV.

Three Eberline Model PING-1A airborne particulate, radiogas, and radioiodine monitor. Radioiodine monitor, usually used with TEDA impregnated charcoal, will also accept the Silver Zeolite cartridge. The detector is a stabilized NaI detector connected to a two channel analyzer calibrated to 365 KeV with automatic Xe subtraction from the second channel.

All cart-mounted iodine detectors are in 3 inch lead shields.

In addition, there are available for use throughout the plant ten Eberline RAS-1 regulated air samplers, which accept either TEDA impregnated charcoal or Silver Zeolite cartridges.

In addition to the equipment normally available in the regular radio-chemistry counting facility, the following analysis equipment is available for analysis of the Silver Zeolite cartridges that might be used in an emergency:

a. A 4" x 4" NaI crystal connected to a Packard 1024 channel MCA is located in the low background counting facility.

b. In the basement assembly area there is a cartridge purge unit consisting of a T-size bottle of dry nitrogen, regulator, cartridge holder, and associated piping to permit purging of Silver Zeolite or charcoal cartridges with dry nitrogen.

c. Located in the basement assembly area is an Eberline MS-2 single channel analyzer, calibrated to 365 KeV, connected to a 2" x 2" NaI crystal in a 2½" lead shield designed for counting in TEDA-charcoal or Silver Zeolite cartridges.

RESPONSE TO ITEM III.D.3.4. CONTROL ROOM HABITABILITY:

The NRC position to assure that control room operators will be adequately protected against accidental release of toxic gases and radiation to operate and/or shut down the plant under design basis accident conditions is presently under review. Our present schedule is to complete the review and evaluation and make the required submittal of information by February 2,1981. Thus, we are requesting relief of the submittal deadline of January 1, 1981 to February 2, 1981 which will allow us sufficient time to complete our review efforts.

ATTACHMENT I.A-1

INDIANA & MICHIGAN ELECTRIC -vs- INPO

STA INITIAL TRAINING PROGRAM

Indiana & Michigan Electric	INPO
1 - Math background assumed per INPO for engineering or physical science degree.	Engineering math - Ordinary Differential Equations - LaPlace Transforms
2 - Reactor Theory - Atomic and Nuclear Physics - Interaction of Radiation with Matter - The Fission Chain Reaction - Neutron Diffusion and Moderation - Nuclear Reactor Theory (including Neutron Multiplication Factors for a Heterogeneous Reflected Thermal Reactor) - The Time-Dependent Reactor a. Reactor Kinetics b. Control Mechanisms c. Reactivity Feedback d. Fission Product Poisoning e. Core Characteristics and Properties During Lifetime - Subcritical Multiplication and 1/M Plots - Estimates Critical Position Calculations (Reactivity Balances) - Shutdown Margin Calculations - Shutdown Cooling Requirements - Reactor Safety Considerations	Reactor Theory - Atomic and Nuclear Physics - Reactor Statics - Two Group Theory - Dynamics, Point Kinetics - Reactivity Feedback
3 - Reactor Chemistry - Corrosion, Reaction Rates	Reactor Chemistry - Inorganic Chemistry - Corrosion, Reaction Rates
4 - Nuclear Metalurgy - Crystallography/Phase Diagrams - Hardening - Response to Stress and Temperature - Properties of Reactor Materials - Zircaloy - Water Reaction - Types of Corrosion	Nuclear Materials - Strength of Materials - Reactor Material Properties - Phase Diagrams - Fuel Densification
5 - Thermodynamics/Fluid Dynamics/Heat Transfer - Laws of Thermodynamics - Equations of State - Steam Cycles/Efficiency - Bernoulli's Equation - Fluid Friction and Head Loss	Thermodynamics/Fluid Dynamics/Heat Transfer - Laws of Thermodynamics - Properties of Water/Steam - Steam Cycles/Efficiency - Bernoulli's Equation - Fluid Friction and Head Loss

Indiana & Michigan Electric	INPO
5 - Theromdynamics/Fluid Dynamics/Heat Transfer (Cont)	Thermodynamics/Fluid Dynamics/Heat Transfer (Cont)
- Compressible Flow	- Elevation Head
- Incompressible Isentropic Flow	- Pump and System Characteristics
- Real Flow Problems	- Two Phase Flow
- Pump Characteristics	
- Two Phase Flow	
- Instrumentation	- Methods of Heat Transfer
- Methods of Heat Transfer	- Boiling Heat Transfer
- Specific Heat, Expansion, Viscosity	- Heat Exchangers
- Viscous Flow	
- Combined Conduction/Convection	
- Nucleate/Film Boiling	
- Critical Heat Flux	
6 - Electrical Sciences	Electrical Sciences
- 4160 Volt Electrical Distribution	- Electronics
- Protective Relaying for Generators	- Motors, Generators, Transformers, Switchgear
- 600 VAC, 120 VAC, and 250 VDC Electrical Distribution	- Instrumentation and Control Theory
- Steam Generator Level Control	
- Feed Pump Speed Control	
- Pressurizer Level/Pressure Control	
- Full Length Rod Control	
7 - Nuclear Instrumentation and Control	Nuclear Instrumentation and Control
- Excore Nuclear Instrumentation System	- Radiation Detectors
- Incore Nuclear Instrumentation System	- Reactor Instrumentation
	- Reactivity Control and Feedback
8 - R/P and Health Physics	Nuclear Radiation Protection and Health Physics
- Site Emergency Plan and Implementing Procedures	- Biological Effects
- Applicable Radiation Protection, Concepts Contained in 10 CFR 20 and 10 CFR 100	- Radiation Survey Instrumentation
- Radiological Control Instructions	- Shielding
- Radiation Monitoring System	
- Portable Radiation Monitoring Instrumentations	
9 - Plant Specific Reactor Technology	Plant Specific Reactor Technology (Including Core Physics Data)
- Reactor Core System	
- Reactor Coolant System	
- Pressurizer and Pressure Relief Systems	
- Full Length Rod Control System	
- Residual Heat Removal System	
- Emergency Core Cooling System	
- Reactor Protection System	

	Indiana & Michigan Electric	INPO
9	- Plant Specific Reactor Technology (Cont)	Plant Specific Reactor Technology
	Plant Chemical and Corrosion Control	- Plant Chemistry and Corrosion Control
	- Chemical and Volume Control and Boron Makeup/Recovery System	
	- Primary and Secondary Sampling System	- Reactor Instrumentation and Control
	Plant Instrumentation and Control	- Reactor, Plant Materials
	-See Item #6, Electrical Sciences, and Item #7, Nuclear Instrumentation and Control	
	Plant Materials	
	- See Item #4, 'Nuclear Metalurgy'	
	Plant Thermocycle	- Reactor Plant Thermocycle
	- Steam, Condensate, and Feed Systems	
	- Steam Dump Systems	
	- Steam Generating and Steam Generator Systems	
10 -	Management/Supervisory	Management/Supervisory Skills
	- to be covered on a selected personal basis after initial STA Training Program	- Leadership
		- Interpersonal Communication
		- Motivation of Personnel
		- Problems and Decisional Analysis
		- Command Responsibility and Limits
		- Stress
		- Human Behavior
11 -	Plant Systems	Plant Systems
	- Reactor Core System	
	- Reactor Coolant System	- Reactor Coolant System
	- Pressurizer and Pressure Relief Systems	
	- Full Length Rod Control System (including Rod Position Indication)	- Reactor Control
	- Chemical & Volume Control System and Boron Makeup and Recovery Systems	- Reactor Coolant Inventory and Chemistry Control
	- Residual Heat Removal System	- Residual Heat Removal System
	- Emergency Core Cooling System	- Emergency Core Cooling System
		- Emergency Cooling Water
	- Excore Nuclear Instrumentation System	- Nuclear Instrumentation
	- Incore Nuclear Instrumentation System	
	- Reactor Protection System	- Reactor Protection System
	- Containment System	- Containment System (including Containment Cooling)
	- Ice Condenser System	
	- Containment Spray and Hydrogen Recombiner System	- Containment Hydrogen Monitoring and Control

Indiana & Michigan Electric	INPO
(Cont) 11- Plant Systems	Plant Systems
- essential service water system	
- non-essential service water system	
- spent fuel pit cooling and cleanup	
- waste disposal system - liquid and gaseous only	
- containment ventilation system	- plant ventilation
- auxiliary building and control room ventilation systems	
- emergency diesel generator system	- emergency electrical power
- auxiliary feedwater system	- auxiliary feedwater system
- compressed air system	- emergency control air
- primary water system	
- primary gas system	
- water fire protection system	
- carbon dioxide fire protection system	
- miscellaneous fire protection system	
- radiation monitoring system	- radiation monitoring system
- portable radiation monitoring instruments	
- steam generating and steam generator water level control systems	- steam generator level control
- steam, condensate and feed systems	- main steam, condensate, and feedwater
- steam dump system	
- 4160 KV electrical distribution system	
- 600 VAC, 120 VAC, and 250 VDC electrical distribution system	
- NOTE: see incore instrumentation and reactor coolant systems	- non nuclear instrumentation
	- loose part monitoring
	- status monitoring (including computer)
	- seismic monitoring
12 - Administrative Control	Administrative Control
- Responsibilities for Safe Operation & Shutdown	- Responsibilities for Safe Operation & Shutdown
- Equipment Outages and Clearance Procedures	- Equipment Outages and Clearance Procedures
- Use of Procedures	- Use of Procedures
- Plant Modifications	- Plant Modifications
- Shift Relief Turnover and Manning	- Shift Relief Turnover and Manning
- Containment Access	- Containment Access
- Maintaining Cognizance of Plant Status	- Maintaining Cognizance of Plant Status
- Unit Interface Controls	- Unit Interface Controls
- Physical Security	- Physical Security
- Control Room Access	- Control Room Access
- Duties and Responsibilities of the STA	- Duties and Responsibilities of the STA
- Radiological Emergency Plan	- Radiological Emergency Plan
- Code of Federal Regulations (appropriate sections)	- Code of Federal Regulations (appropriate sections)

Indiana & Michigan Electric	INPO
(Cont) 12 - Administrative Controls - Plant Technical Specifications (including bases) - Radiological Control Instructions	Administrative Controls - Plant Technical Specifications (including bases) - Radiological Control Instructions
13 - Transient and Accident Analysis for Shift Technical Advisors with reference to the FSAR and Plant Abnormal and Emergency Procedures	Transient and Accident Analysis for Shift Technical Advisors with reference to the FSAR and Plant Abnormal and Emergency Procedures
14 - Normal, Transient, and Emergency Operations (Simulator Training)	Normal, Transient, and Emergency Operations (Simulator Training)
15 - General Operating Procedures - NOTE: covered in systems and reactor theory lectures	General Operating Procedures - Startup - At power operations - Shutdown - Xenon following while on standby - ECP and S.D. margin calculation

ATTACHMENT I.A-2

STA REQUALIFICATION PROGRAM

The Shift Technical Advisor Requalification Program shall be conducted on an annual cycle basis.

The requalification program shall consist of:

1. Formal classroom lectures
2. On-the-job training (including simulator training)
3. An annual evaluation
4. Training documentation.

Lectures shall be conducted in the following areas with emphasis on identified weak or problem areas:

a. Theory and Principles of Operation (includes Thermodynamics, Heat Transfer and Fluid Flow)
b. General and Specific Plant Operating Characteristics
c. Plant Instrumentation and Control Systems
d. Plant Protection Systems
e. Engineered Safety Systems
f. Normal, Abnormal and Emergency Operating Procedures
g. Radiation Control and Safety
h. Technical Specifications
i. Applicable portions of Title 10, Chapter 1, Code of Federal Regulations

The use of training aids such as videotapes or films may be used in lieu of an instructor. However, no more than 50% of the lecture series shall be solely videotape or film.

The annual lecture series will be of an estimated length of 40 hours, but in no case less than 30 hours. Lectures shall be evenly spaced throughout the period, taking infrequent operations such as refueling operations into account.

Written quizzes will be administered after each lecture topic for the evaluation of individual knowledge level and progress.

On-the-job training shall consist of:

a. Supervision and/or performance of control manipulations (simulator training)
b. On-shift abnormal and emergency procedure review
c. Keeping abreast of all facility and procedure changes
d. Review of NRC, vendor, and industry related notices.

Each shift technical advisor shall, during the requalification training cycle, perform/supervise a minimum of plant control manipulations which demonstrate his skill and/or familiarity with plant control systems.

Each shift technical advisor shall either direct or evaluate the activities of others or manipulate the controls during these control manipulations. It shall be emphasized that the shift technical advisor serve in his designated role during these control manipulations where possible.

As many of the following control manipulations as possible should be performed during each requalification cycle. The asterisked items shall be performed annually.

- *1. Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.
- 2. Plant shutdown
- *3. Manual control of steam generators and/or feedwater during startup or shutdown.
- 4. Boration and/or dilution during power operation
- *5. Any significant (>10%) power changes in manual rod control
- *6. Loss of coolant, including:
  1. Significant steam generator leaks.
  2. Inside and Outside primary containment.
  3. Large and small, including leak-rate determination.
  4. Saturated Reactor Coolant response.
- 7. Loss of electrical power (and/or degraded power sources).
- *8. Loss of core coolant flow/natural circulation.
- 9. Loss of condenser vacuum.
- 10. Loss of Essential Service Water.
- *11. Loss of shutdown cooling.
- 12. Loss of Component Cooling System or cooling to an individual component.
- 13. Loss of normal feedwater or normal Feedwater System failure.
- *14. Loss of all feedwater (normal and emergency).
- 15. Loss of protective system channel.
- 16. Mispositioned control rod(s) (or rod drops).
- 17. Inability to drive control rods.
- 18. Conditions requiring use of emergency boration.
- *19. Fuel cladding failure or high activity in reactor coolant or offgas.
- 20. Turbine or generator trip.
- 21. Malfunction of Automatic Control System(s) which affect reactivity.
- 22. Malfunction of Reactor Coolant Pressure/Volume Control System.
- *23. Reactor Trip.
- *24. Main steam line break (inside or outside containment).
- 25. Nuclear Instrumentation failure(s).

* Required at least annually.

Even if the above manipulations are not needed to be accomplished by a simulator, each shift technical advisor shall attend a training session at an appropriate simulator annually.

Abnormal and emergency procedures shall be reviewed by all shift technical advisors on a regularly scheduled basis as assigned by the Training Coordinator. The procedure review shall normally be accomplished each shift cycle by on-shift selfstudy. Other areas of interest may be included in the periodic review assignment. All abnormal and emergency plant operating procedures shall be reviewed at least annually.

All shift technical advisors shall review on a continuous basis all changes in facility design, operating procedures and the facility license. The determination of the depth of review of any changes shall be made by the Training Coordinator or cognizant Department Head. Reviews shall be conducted by one of the following methods:

1. Formal training lectures, to be scheduled and conducted during requalification lectures.
2. Individual review, to be read by the individual during his normal work hours. Questions to be directed to the Training Department.
3. Shift group discussion, to be conducted on-shift by the Shift Operating Engineer.

All shift technical advisors shall receive a written examination annually to determine the effectiveness of the overall requalification program and to define those areas where additional emphasis is required. An overall grade average of less than 80% or any category grade of less than 70% shall require the individual to be placed on an accelerated training program prepared to correct the identified weakness. The scope and duration of the accelerated training program shall be based upon management evaluation in each instance it is required.

A permanent record shall be maintained for each shift technical advisor containing verification of each program completion and the overall grade scores for the annual written examination. This permanent record file shall be maintained for the life of the facility.

ATTACHMENT II.E-1

Information for implementating positions 6 and 7 of item II.E.4.2 of NUREG-0737 in Cook Plant.

1. Our response to Questions 022.4 and 022.13 contained in Appendix Q, Amendments 77 and 78 to the FSAR, submitted in July and October 1977 respectively. This provided our response to Branch Technical Position CSB 6-4 including the impact on ECCS performance and an evaluation of the radiological consequences of a design basis accident during purge operation. The response to Question 022.4 and 022.13 apply to both Units of the Cook Plant.

2. Our submittal of December 29, 1977 on Unit 2 provided the procedure for the valve operability test. The scope of this test procedure was reviewed with members of your staff prior to our submittal. This applies to both Units 1 and 2 of the Cook Nuclear Plant.

3. Our submittal of January 13, 1978 on Unit 2 provided the results of the in-situ purge valve operability test performed on January 8, 1978. This test was a pre-requisite for allowing unrestricted purging of the containment in accordance with our response to Containment Systems Branch Question 022.4 of our FSAR. The test results demonstrated that the purge valves are capable of closing against the dynamic forces of a design basis loss-of-coolant accident. These results were submitted to the Commission in support of our Technical Specification change request on Unit 2 to allow unrestricted purging of the containment. This test, its results and the various supporting analyses we have performed address the concerns expressed in Mr. Eisenhut's letter of September 27, 1979 and no further action is required. This applies to both Units 1 and 2 of the Cook Plant.

4. Our submittal of February 3, 1978 on Unit 2 provided supplemental information requested by your staff concerning the results of the valve operability test in support of our Technical Specification change request. This applies to both Units 1 and 2 of the Cook Plant.

5. Our submittal of April 27, 1978 on Unit 2 supplied analyses that demonstrate the operability of the lower compartment purge system based on the test already performed. The analysis provided shows that although lower compartment pressures might be higher than the test pressure, the pressures expected at the inboard containment isolation valves in the lower compartment purge and vent lines would be less than the pressure which existed during the valve operability test. This is achieved by installing debris screens in the lower compartment purge systems which provide a high flow resistance. This applies to both Units 1 and 2 of the Cook Plant.

6. Our submittal of August 11, 1978 (AEP:NRC:00069) on Unit 2 provided additional information requested by your staff and applies to both Units of the Cook Plant.

7. Our submittal of September 11, 1978 (AEP:NRC:00082) on Unit 2 provided sensitivity analyses of the resistance coefficients for the elbows and debris screens and the dependence on those coefficients of the resulting torque and applies to both Units 1 and 2 of the Cook Plant.

8. Our submittal of January 4, 1979 (AEP:NRC:00114) on Units 1 and 2 provide our response to Mr. Schwencer's November 28, 1978 letter. All of the requests for additional information and justification of unlimited purging were provided. Additionally, we provided our review of the issue of overriding of safety actuation signals and the procedural steps taken to assure that operation of a bypass will not affect safety functions.

9. Our meeting with the NRC staff on May 31, 1979 to discuss the status of review of the containment purge and related subjects. The NRC staff informed us that we had a favorable writeoff as far as valve operability was concerned. However, we were told that the NRC staff required further action from AEP on the issue of manual override of safety actuation signals.

10. Our submittal of June 8, 1979 (AEP:NRC:00114A), applicable to Units 1 and 2 provided a descriptions of the modifications made to the reset/ block circuits and associated procedural changes required to meet the Commission's position as committed to at the May 31, 1979 meeting.

11. Our submittal of June 29, 1979 (AEP:NRC:00114B) on both Units 1 and 2 provided the additional information on the subject of unrestricted purging that was requested at the May 31, 1979 meeting.

12. Our submittal of November 8, 1979 (AEP:NRC:00295) provided our response to Mr. D. Eisenhut's letter of September 27, 1979 which dealt with containment purging and venting during normal operation. This submittal also provided the completion of our review of overriding of safety actuation signals.

13. Our submittal of December 5, 1979 (AEP:NRC:00295A) provided further information requested by the NRC staff concerning the monitoring of containment radiation in the context of the purge system operation.

14. Our submittal of March 10, 1980 (AEP:NRC:00370) provided our response to Mr. D. Eisenhut's letter of February 11, 1980 regarding the Commission's interim position concerning containment purging and venting during normal plant operation.

15. Our submittal of March 25, 1980 (AEP:NRC:00295B) provided further information requested by the NRC staff concerning the control room isolation function and containment radiation monitors as a result of the on-going review of containment purging matters.

UNIT 1

TABLE II.K.3 - 1
D.C.COOK UNIT 1 ECC SYSTEM OUTAGES(II.K.3.17)

Date/Time	Duration	Equipment	Reason
1-18-75		Initial Criticality	
2-25-75	57 hrs	"AB" Diesel Engine	Lubrication oil piping modifications
3-3-75	62 hrs	"CD" Diesel Engine	Lube oil piping modifications
1-13-75	36 hrs*	RHR System	leak
4-7-77	57 hrs	West Centrifugal Charging Pump	Pump shaft severed
5-17-77	67 hrs	East Centrifugal Charging Pump	Broken shaft
8-9-77	36 hrs*	East Centrifugal Charging Pump	leak repair
8-11-77	9 hrs	"AB" Diesel Generator	wrist pin replacement
8-30-77	36 hrs*	West RHR	motor modification
9-12-77	11 hrs	"CD" Diesel	wrist pin replacement
9-14-77	36 hrs*	"CD" Diesel	maintenance
2-6-78	5 hrs	"AB" Emergency Diesel	tach. modification
2-7-78	3.3 hrs	"CD" Emergency Diesel	tach. modification
9-7-78	3.5 hrs	North SI Pump	maintenance
9-7-78	3 hrs	South SI Pump	maintenance
10-3-78	15 hrs	East CTS Pump	valve repair
11-7-78	4.25 hrs	"AB" Emergency Diesel	replace voltage regulator
11-10-78	8 hrs	"CD" Emergency Diesel	maintenance
11-18-78	5 hrs	"AB" Emergency Diesel	maintenance
11-25-78	8 hrs	East Centrifugal Charging Pump	oil leak
11-30-78	12 hrs	East RHR System	clearance on IMO-312
12-6-78	8.5 hrs	"CD" Emergency Diesel	design change
1-17-79	36 hrs*	West RHR Train	repair RH128W
1-26-79	9.5 hrs	"AB" Emergency Diesel Generator	design change
1-30-79	33.5 hrs	East Centrifugal Charging Pump	seal failure
2-15-79	6 hrs	East Centrifugal Charging Pump	maintenance
2-15-79	5.7 hrs	West Centrifugal Charging Pump	change oil strainer
2-18-79	3.4 hrs	West Centrifugal Charging Pump	repair QPI257 isolation vlv.

UNIT 1

TABLE II.K.3 - 1

Date/Time	Duration	Equipment	Reason
2/27-2/28/79	2.2 hrs	East and West RHR Pumps	brkr inspection
2/27-2/28/79	2.5 hrs	North and South SI Pumps	brkr inspection
2/27-2/28/79	1.2 hrs	East and West CTS Pump	brkr cleaning
2-23-79	7.6 hrs	"AB" Diesel	maintenance on ESW check vlvs.
3-(1-7)-79	2.2 hrs	West Centrifugal Charging Pump	clean brkr.
3-1-79	11.2 hrs	"CD" Diesel	CD-work on ESW ck.vlv.
3-3-79	6.0 hrs	"AB" Diesel	AB-work on ESW ck.vlv.
3-12-79	36 hrs*	"CD" Emergency Diesel Generator	maintenance
3-16-79	36 hrs*	"AB" Emergency Diesel Generator	maintenance
4-(5-10)-79	36 hrs*	"AB" Emergency Diesel	maintenance
7-23-79	5 hrs	North SI Pump	repack ICM-250
8-1-79	2.6 hrs	West Centrifugal Charging Pump	maintenance
9-24-79	4.5 hrs	South SI Pump	maintenance
9-25-79	1 hr	North SI Pump	maintenance
9-25-79	2 hrs	East RHR Pump	maintenance
9-28-79	2 hrs	West RHR Pump	maintenance
1-30-80	6.5 hrs	"AB" Diesel	Performance testing
1-31-80	3.4 hrs	"CD" Diesel	replace relays - RFC
2-26-80	6 hrs	"AB" Emergency Diesel	maintenance
3-5-80	1.5 hrs	East Centrifugal Charging Pump	maintenance clean MCC
3-6-80	30.75 hrs	East Centrifugal Charging Pump	change rotating assembly
3-26-80	5.2 hrs	East Centrifugal Charging Pump	clean CONO filter on lube oil system
3-30-80	3.5 hrs	East Containment Spray Pump	maintenance
3-21-80	4.1 hrs	South Safety Injection Pump	clean brkr.
3-24-80	2.1 hrs	West RHR Pump	clean brkr.
	2.7 hrs	West Containment Spray Pump	clean brkr.
	1.6 hrs	West Centrifugal Charging Pump	clean brkr.

UNIT 1

TABLE II.K.3 - 1

Date/Time	Duration	Equipment	Reason
3-25-80	2 hrs	East Centrifugal Charging Pump	clean brkr.
3-26-80	9.0 hrs	"CD" Emergency Diesel Generator	clean output brkrs.
3-27-80	9.3 hrs	"AB" Emergency Diesel	clean output brkrs.
4-19-80	1.75 hrs	East Centrifugal Charging Pump	change oil

UNIT 2.

TABLE II.K.3 - 2
D.C.COOK UNIT 2 ECC SYSTEM OUTAGES(II.K.3.17)

Date/Time	Duration	Equipment	Reason
3-10-78		Initial Criticality	
4-16-78	10 hrs.	West RHR Train	Erroneously declared inoperable (RH-104W Pos. Indication at mid-position.
5-15-78	39 hrs	"CD" Emergency Diesel	repair water leak on Air Cooler.
7-11-78	11.7 hrs	"AB" Emergency Diesel	replace ESW Supply Spool Piece
7-18-78	9.5 hrs	"CD" Emergency Diesel	repair fuel injection
9-11-78	8.7 hrs	"AB" Emergency Diesel	Starting Air Ck. Vlv. leaking through
9-12-78	49.5 hrs	"CD" Emergency Diesel	fuel oil leaks
9-19-78	2 hrs	West RHR Pump	Repair brkr. leakage
10-3-78	36 hrs*	East CTS Train	Valve Problem could not verify flow through CTS-120E - Broken shaft IMO-21.
10-11-78	16 hrs	"CD" Emergency Diesel	routine maintenance
10-19-78	0.5 hrs	"AB" Diesel	Failed to start twice.
10-26-78	1.75 hrs	"CD" Emergency	Relay Calibration
11-17-78	18.9 hrs	"AB" Emergency Diesel Generator	routine maintenance
1-11-79	4.2 hrs	E RHR Pump	lubrication
1-11-79	2.1 hrs	W RHR Pump	lubrication
1-12-79	8.3 hrs	East Centrifugal Charging Pump	oil change
1-24-79	17.3 hrs	"AB" Diesel Generator	Repair ESW Supply Vlv.
1-31-79	7.7 hrs	"AB" Emergency Diesel Generator	Repair ESW-143
4-23-79	21 hrs	"CD" Emergency Diesel Generator	Repair ESW Supply ck. vlv.
7-18-79	7.5 hrs	"CD" Emergency Diesel Generator	Repair leaking Ck. Vlv.
8-27-79	8 min	"AB" Emergency Diesel	Testing Co_2 to Emer. Diesel Rooms.
8-27-79	8 min.	"CD" Emergency Diesel	Testing Co_2 to Emer. Diesel Rooms

UNIT 2

TABLE II.K.3 - 2

Date/Time	Duration	Equipment	Reason
9-26-79	36 hrs*	East RHR Pump	routine maintenance
9-29-79	7 hrs	West Centrifugal Charging Pump	repack coupling
10-15-79	9.1 hrs	"CD" Emergency Diesel Generator	routine inspection
10-16-79	11 hrs	"AB" Emergency Diesel Generator	routine inspection
1-18-80	8 hrs	East Containment Spray Pump	remove suction strainer
1-21-80	13 hrs	West Containment Spray Pump	remove suction strainer
1-30-80	7.2 hrs	North SI Pump	remove suction strainer
1-31-80	7.3 hrs	South SI Pump	remove suction strainer
2-1-80	6 hrs	West Centrifugal Charging Pump	check suction strainer
2-4-80	2.0 hrs	North SI Pump	Repair SV-98 on Discharge Header
2-4-80	12.5 hrs	East RHR Heat Exchanger	Repair SV-104
2-11-80	39.2 hrs	West RHR Pump & Heat Exchanger	Repack IMO-324
2-14-80	9.2 hrs	West RHR Pump	Repack IMO-322
2-28-80	7.9 hrs	East Centrifugal Charging Pump	remove suction strainer
3-17-80	7.1 hrs	West Centrifugal Charging Pump	Fitting leak oil supply line
4-1-80	10 hrs	West Centrifugal Charging Pump	Replace gasket on suction strainer
4-10-80	57.5 hrs	West Centrifugal Charging Pump	Seal Failure
4-15-80	5.9 hrs	West Centrifugal Charging Pump	Repair leak on CCW, to Heat Exch.
4-16-80	1.5 hrs	West Centrifugal Charging Pump	Output brkr. maint.