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DENTON,H.R.	Office of Nuclear Reactor Regulation, Director

SUBJECT: Responds to Generic Ltr 82-33 re Suppl 1 to NUREG-0737, "Requirements for Emergency Response Capability." Current status/schedules & integrated plans encl.

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INDIANA & MICHIGAN ELECTRIC COMPANY

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NEW YORK, N. Y. 10004

April 15, 1983
AEP:NRC:0773

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
GENERIC LETTER NO. 82-33
SUPPLEMENT 1 TO NUREG-0737
REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY

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

Dear Mr. Denton:

On January 3, 1983, we received NRC Generic Letter No. 82-33, Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability", dated December 17, 1982.

We have reviewed Generic Letter 82-33 and we have determined that the items identified in the enclosures to the letter which our response must address, are:

a. Safety Parameter Display System (SPDS)
b. Detailed Control Room Design Review (DCRDR)
c. Regulatory Guide 1.97 - Application to Emergency Response Facilities
d. Upgrade Emergency Operating Procedures (EOPs)
e. Emergency Response Facilities (ERFs)

We have addressed requirements for two of the above items in earlier submittals to the NRC prior to the issuance of Supplement 1 to NUREG-0737, within the context of our TMI-2/NUREG-0737 responses. The two items are (a) the Safety Parameter Display System (SPDS) and (e) the Emergency Response Facilities (ERFs). Our early efforts show compliance between our SPDS and ERFs and the specific final guidance issued by the NRC in Supplement 1 to NUREG-0737. We do not know whether differences may arise between our implementation of these two items and Supplement 1 to NUREG-0737 as our activities proceed on the remaining three items above, (b), (c), and (d). We will rely on the

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Add: W. Paulson

NRC commitment in Supplement 1 to NUREG-0737, Section 3.7, to make allowances for the work already done in a good-faith effort.

Generic Letter No. 82-33 requests us to furnish to the NRC, by April 15, 1983, a proposed schedule for completing each of the basic requirements for the items identified in the enclosures to the letter, and with it a description of our plans for phased implementation and integration of the emergency response activities.

Attachment 1 to this letter is a description of the current status of the major items addressed in Generic Letter 82-33.

Attachment 2 to this letter is a description of our Plan for Phased Implementation and Integration of the Emergency Response Activities. This Plan takes into consideration the work already completed on the Safety Parameter Display System and on the Emergency Response Facilities. The completed work on these two items (as discussed in Attachment 1) was taken as a "given" in the development of our Plan.

Attachment 3 to this letter summarizes the expected dates of completion for those activities which we believe can be planned at this time. For the reasons outlined in Attachment 2, it is not possible at this time to commit to firm schedules for final implementation of all the NRC requirements in Supplement 1 to NUREG-0737. Rather, we are providing with this letter realistic and achievable dates for certain elements of the Plan, and an intermediate milestone in place of an end date. The intermediate milestone is the date by which necessary additional information will be available or work progress will permit a commitment to a final completion date.

We were advised at the NRC Region III Workshop held on March 4, 1983 that referencing intermediate milestones instead of supplying explicit end dates is acceptable to the NRC.

To ensure timely notification to the NRC of the status of our Plan, we will provide the NRC with another letter by December 31, 1983. These schedules will require integration with all other NRC required Plant modifications and with the scheduled outages for Units 1 and 2.

As Generic Letter No. 82-33 encourages, we are prepared to meet with our NRC Project Manager to explain our integrated implementation plan in more detail, so that we can reach an agreement on a mutually acceptable schedule.

Very truly yours,

R. F. Hering

R. F. Hering
Vice President

RFH/emc
Attachments

cc: John E. Dolan - Columbus
M. P. Alexich
R. W. Jurgensen
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
NRC Resident Inspector at Cook Plant - Bridgman

STATE OF NEW YORK)
)
COUNTY OF NEW YORK)

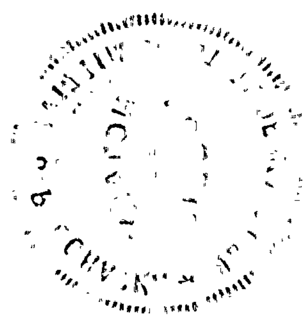
R. F. Hering, being duly sworn, deposes and says that he is the Vice President of Licensee Indiana & Michigan Electric Company, that he has read the foregoing response to Generic Letter No. 82-33, and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

R. F. Hering

Subscribed and sworn to before me this 15th day of April, 1983.

William J. Prochaska
(Notary Public)

WILLIAM J. PROCHASKA
Notary Public, State of New York
No. 43-4636690
Qualified in Richmond County
Certificate filed in New York County
Commission Expires March 30, 1984



ATTACHMENT 1 TO AEP:NRC:0773
CURRENT STATUS OF THE
MAJOR ITEMS IN GENERIC LETTER 82-33

The following paragraphs describe the current status (and background information where appropriate) of the major items addressed in Generic Letter 82-33.

BACKGROUND

On September 13, 1979, the NRC transmitted to us the report titled "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578). One of the requirements set forth in this report was that, "each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident".

Our response (letter No. AEP:NRC:00253, dated October 24, 1979) stated that the Westinghouse TMI Owners' Group was reviewing the basis for display of Plant data, methods of displaying data and investigating communications needs.

On March 27, 1980, members of the American Electric Power Service Corporation engineering staff, Westinghouse Electric Corporation, and representatives of two other utilities met with the NRC Staff in Bethesda, Maryland to inform the NRC of our Technical Support Center (TSC) approach, to receive feedback on function and objectives, and to lay a foundation for future review meetings and schedulings. Westinghouse made a presentation describing the TSC as an integrated system which incorporates the following three functions:

(1) Onsite Technical Support Center (OTSC) - Provide key Plant information in a location separated from the control room to support post-accident recovery management. This center also provides the means by which Plant operating data can be transmitted to offsite locations.

(2) Bypassed and Inoperable Status Indication (BISI) - Provide the operator with a clear indication of the availability status of certain Plant safety systems.

(3) Plant Safety Status Display (PSSD) - Provide the operator with a succinct account of the Plant status as it pertains to Plant safety.

Reaction from the NRC Staff was favorable. The Staff agreed that the Westinghouse TSC concept exceeded minimum requirements. No feedback was received on the Westinghouse proposed licensing approach which was to hold five generic meetings followed by a WCAP to be submitted in mid-1980. On May 22, 1980, Indiana & Michigan Electric Company issued a purchase order to Westinghouse for their TSC Complex to gain an early start in the design, testing and installation of the Westinghouse-designed TSC data display systems.

On May 27, 1980, American Electric Power Service Corporation engineering management and Westinghouse personnel held a second meeting with the NRC Staff (Messrs. S. Hanauer, R. Mattson, S. Miner, W. Minners, Beltracchi, and V. Moore) specifically on the Cook Plant TSC Complex. AEP described a data collection and evaluation system (DCES) which it had designed and operated in its coal-fired power plants. The DCES experience was a factor in going forward with the Westinghouse TSC concept. Westinghouse gave a technical presentation on the TSC Complex and concluded with a proposed licensing submittal. The NRC Staff indicated that they could not meet the proposed Westinghouse schedule for NRC review of a WCAP in the June 16 to August 1, 1980 time period. The NRC Staff stated that they preferred working with an AIF Subcommittee on Safety Parameter Integration which had been formed to provide industry comment to the NRC, and they asked us to join with the AIF. Final NRC criteria were planned to be issued by the end of June, 1980, followed by more detailed criteria for the Safety Parameter Display System (SPDS) in September, 1980. As a result of the meeting, AEPSC did assign a member to the AIF Subcommittee on Safety Parameter Integration. Simultaneously, AEPSC and Westinghouse proceeded to design and build the TSC Complex described to the NRC at these meetings. Westinghouse submitted Report No. WCAP-9725 in 1980. This report describes the TSC computer system, the technical basis for parameter selection for the SPDS and TSC systems, and its functions and interface with Plant instrumentation.

On March 12, 1981, we received a copy of NRC Generic Letter No. 17 which enclosed a copy of NUREG-0696, "Functional Criteria for Emergency Response Facilities (Final Report)".

On June 19, 1981, we transmitted the "Facility Conceptual Design Description for the Cook Plant TSC" as an Attachment to our letter No. AEP:NRC:0531A. The Report presented a TSC parameters list, the design basis, and the data display system description specifically for the Cook Plant based on Westinghouse's WCAP-9725.

(a) SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

CURRENT STATUS

The acronym "SPDS" is equal in meaning to the Westinghouse PSSD referenced above. SPDS design is complete except for physical wiring diagrams for some supplementary input signals. Only those revisions found to be required to bring the system function up to specification as a result of Plant operational testing are planned for the future.

SPDS OPERATIONAL AND TRAINING DATES

SPDS hardware for Unit 1 has been installed and checked out. The majority of the Unit 2 SPDS hardware has been installed and checked out, however, certain items such as fixed head disks were retained by Westinghouse for software development. SPDS inputs that meet the criteria of our submittal documentation and are specified by Regulatory Guide 1.97, Rev. 2, have been provided to the TSC Computers (which drive our SPDS displays). Other inputs are to be provided as Plant operating and manpower conditions allow.

The TSC computers are presently shutdown to allow for TSC facility and hardware upgrade. We are also awaiting delivery of final verified software (SPDS, TSC, EOF and BISI) from Westinghouse, which is expected before October 1, 1983. At that time, formal operator training will commence. (Some operators are already familiar with the display as a result of the operation of previous versions of SPDS software on the Unit 1 TSC Computer.)

SPDS VERIFICATION AND VALIDATION

As originally specified, our SPDS called for no special documentation for NRC review. This system is non-Class 1E which was purchased and installed under Standard (non-nuclear safety) Quality Assurance Procedures. This approach was a good faith effort to meet SPDS requirements as they existed at the time of the original SPDS design work. As such, we do not plan on the submission of documentation or procedures for NRC review other than those to which we have already committed to in past correspondence such as the report provided by AEP:NRC:0531A.

(b) DETAILED CONTROL ROOM DESIGN REVIEW (DCRDR)

CURRENT STATUS

We have developed a specification which defines the technical requirements of I&MECo.'s DCRDR Program. This document, which is available for your review, has been formally issued to outside consulting engineering firms with expertise in human factors,

soliciting their proposals to provide the necessary engineering, administrative, and program management services to plan, perform and document the DCRDR Program as specified by AEPSC. The proposal evaluation has been completed by AEPSC, and Torrey Pines Technology Inc. (TPT) has been selected as the contractor for the DCRDR Program based on their ability to meet AEPSC's requirements. In a combined effort, AEPSC and TPT will develop the DCRDR Program Plan Report (PPR) for submittal to the NRC as required by Supplement 1 to NUREG-0737 (Generic Letter 82-33), Item 5.2.a.

In conceptual agreement with the organizational guidance provided in NUREG-0700, Section 1.4 "The Control Room Design Review Process" the four major activities in our DCRDR Program are identified as:

Phase I - Planning
Phase II - Review
Phase III - Assessment and Implementation
Phase IV - Reporting

Descriptions of each phase will be presented in our PPR.

(c) REGULATORY GUIDE 1.97 - APPLICATION TO EMERGENCY RESPONSE FACILITIES

CURRENT STATUS

During the engineering and design of the Technical Support Center Computer system, the parameter listing for Regulatory Guide 1.97, Rev. 2 was reviewed. In addition, during the various submittals made to the NRC relating to the Equipment Qualification Bulletin IEB 79-01B, the parameters of this regulatory guide were considered for qualification of equipment. Therefore, it can be stated that a monitoring device, either indicator, recorders, CRT display, and/or alarm, is available in the control room and/or system subpanel for each of the Regulatory Guide 1.97, Rev. 2 parameters.

In addition, the analog parameters are from existing Plant instruments through appropriate circuit isolation by either normal Plant design or by the method used during the TSC Computer System design.

(d) UPGRADE EMERGENCY OPERATING PROCEDURES (EOPs)

BACKGROUND

The primary objective of the EOP Update program is to bring the D. C. Cook Nuclear Plant Operating Procedures into compliance with the requirements of NUREG-0737, Item I.C.1. Compliance is to be obtained by upgrading the existing Plant EOPs and implementation of these

upgraded EOPs on a schedule mutually agreed upon by I&MECo. and the NRC.

The upgraded EOPs produced for the D. C. Cook Nuclear Plant will utilize the generic Westinghouse Owners' Group Emergency Response Guidelines (ERGs) as a basis. Changes to the technical background of the ERGs will be made only where design differences between the D. C. Cook Nuclear Plant and the reference configuration NSSS assumed for the ERG development make these necessary.

Revision 0 to the Westinghouse generic ERGs was transmitted to the NRC on November 30, 1981, letter No. OG-64. The Function Restoration Guidelines were transmitted by the Westinghouse Owners' Group to the NRC in letter No. OG-83, dated January 4, 1983. Revision 1 to these generic ERGs is under development by the Westinghouse Owners' Group and currently scheduled for completion by July 31, 1983. This revision will address the differences in high and low head safety injection plants, the verification and validation tests performed on the Callaway simulator, and changes resulting from the WOG Procedures Subcommittee reviews. Revision 1 will be used to develop the full set of plant-specific EOPs for Cook Plant.

In support of the implementation of improved EOPs based upon the generic guidelines developed in response to NUREG-0737, Item I.C.1, a joint INPO/Industry program to identify the crucial elements of a plant-specific implementation program, and to provide implementation guidance on a generic basis (where appropriate), is currently in place. This Emergency Operating Procedures Implementation Assistance (EOPIA) program is intended to support a coherent, acceptable industry approach to EOP implementation. EOPIA program guidance was issued in April, 1982. The Westinghouse Owners' Group is now developing generic guidance on creation of plant-specific procedure writer's guides, and validation/verification guidance for plant-specific procedures. Westinghouse is participating in the EOPIA development program through the invitation of the Westinghouse Owners' Group Procedures Subcommittee Chairman, and will continue to be represented throughout the development cycle.

NRC requirements for implementation of EOPs, based upon the post-TMI technical guidelines are contained in NUREG-0899 (successor to NUREG-0799). This document requires the submittal of a Plant Procedures Generation Package (PGP) to the NRC at least three months prior to initial formal operator training on the new EOPs. The PGP is required to contain a Plant-Specific Writer's Guide, a description of the planned method for developing EOPs from the generic technical guidelines (or a set of plant-specific technical guidelines), a description of the validation program for the new EOPs, and a description of the training program to be used to familiarize the operators with the new EOPs.

On August 31, 1982, I&MECo. issued a purchase order to Westinghouse Electric Corporation to perform Phase I of a two-phase EOP upgrade program. The first phase of the program is to develop all plant-specific information necessary for modification and upgrade of the existing plant emergency operating procedures. Phase I, which is in the final stages of completion, consists of four tasks as outlined below:

<u>TASK 1</u> - Review of Existing D. C. Cook EOPs and Definition of Program Scope.

The existing EOPs were reviewed to identify their corresponding guidelines in the ERG set and to define precisely the extent of the upgrading program scope. The interfaces with the remaining Plant procedures were identified.

<u>TASK 2</u> - Specification of Applicable Generic Guidance.

Westinghouse will recommend a set of generic guidance materials for the EOP upgrade program. The nucleus of this data will be NUREG-0899, the INPO EOPIA guidance, and the NRC Generic Letter 82-33. The specific application of each piece of generic guidance to the program will be defined.

<u>TASK 3</u> - Identification and Development of Plant-Specific Information Required for EOP Development.

In most cases, the technical (analytical) basis of the ERGs will be directly comparable to the plant-specific technical basis for the D. C. Cook Nuclear Plant, since the reference NSSS configuration for the ERGs is a standard Westinghouse 4-loop 3425 MWth Plant. However, certain parts of the implemented ERGs may require plant-specific analyses.

<u>TASK 4</u> - Definition of Plan to Develop the D. C. Cook EOPs from the ERGs.

This plan is one of the major elements of the Procedures Generation Package required by the NUREG-0899. Westinghouse is developing a plan and accompanying schedules for the Phase II work. After review by I&MECo. and modification as required, this plan will be used to direct the effort of procedure writing.

On March 14, 1983, I&MECo. requested an updated proposal from Westinghouse for Phase II of the program which involves EOP writing assistance, and review of the completed EOPs, plant-specific writer's guide and the validation program.

(e) EMERGENCY RESPONSE FACILITIES

CURRENT STATUS

EMERGENCY OPERATIONS FACILITY (EOF) Construction of the EOF was completed in October 1982. The EOF is functional with the exception of some data display terminals planned to be made operational as discussed below. The EOF is located approximately 11 miles north of the Cook Plant in Benton Township, Michigan. A description of the EOF was documented to the NRC in the Donald C. Cook Nuclear Plant Emergency Plan, Rev. 1 which was transmitted to the NRC on August 12, 1982 via letter No. AEP:NRC:0719.

The EOF is equipped with all documents required for its function. Sufficient working area has been provided for I&MECo., AEPSC, Federal, State and Local Officials. However, to date, the State and Local officials have chosen not to work out of the EOF, but rather to be in their own Emergency Operating Centers (i.e., St. Joseph for Local Authorities and Lansing for State Authorities).

A successful emergency exercise was completed on October 21, 1982, and accepted by FEMA and the NRC in IE Inspection Reports No. 50-315/82-18 and No. 50-316/82-18, where ten exercise weaknesses were identified. I&MECo. has addressed the weaknesses as discussed in our letter AEP:NRC:0754, submitted to the NRC on December 13, 1982.

Emergency Plan Implementing Procedures have been written and were transmitted to the NRC in our Letters AEP:NRC:0308G, dated March 27, 1981 and AEP:NRC:0777, dated February 1, 1983.

A study is currently being conducted to determine the best location for a back-up power supply (diesel generator) for this facility. All necessary communication links are installed and operational.

There are four data display terminals planned to be located in the EOF. The status of the four display terminals for the EOF is as follows:

1) CPM002 - In-house Dose Assessment Computer Program - installed and functional. (See Meteorological Data Section below)

2) MIDAS - Meteorological Information and Data Acquisition System - installed and functional. (See Meteorological Data Section below)

3) Remote Plant Safety System Display - We expect delivery of the final TSC software package, from Westinghouse, which will include the software to drive this remote terminal, by October 1, 1983. Equipment checkout and testing of this remote terminal will commence at that time. While no firm date exists at this time for operational status of this terminal, we feel that barring the necessity for a complete rewrite of the software by the vendor, this terminal can be made fully operational by September 1, 1984.

4) RDDS - Radiation Data Display System - All equipment, with the exception of a hard-copy display printer, is on site. Installation has been delayed by safety review of the proposed installation's impact on the Plant Radiation Monitoring System. Assuming that the system and its software functions as specified, we plan to have it installed and operating by November 1, 1983.

<u>TECHNICAL SUPPORT CENTER (TSC)</u> Construction of the TSC, including the 5-man NRC office was completed in October 1982. As with the EOF, the TSC is functional with the exception of some data display terminals planned to be made operational as discussed below. The TSC is located adjacent to the control rooms in the Auxiliary Building of the Plant. A description of the TSC was documented to the NRC in the Donald C. Cook Nuclear Plant Emergency Plan, Rev. 1.

All required documents are readily available to personnel working in the TSC. The redundant Heating, Ventilation and Air Conditioning System (HVAC) will be installed and fully operational by September 1, 1983. All necessary communication links are installed and operational.

There are four data display terminals planned to be located in the TSC. The status of these display terminals for the TSC is as follows:

1) CPM002 - In-house Dose Assessment Computer Program - installed and functional. (See Meteorological Data Section below)

2) MIDAS - Meteorological Information and Data Acquisition System - installed and functional. (See Meteorological Data Section below)

3) Westinghouse TSC Displays (OTSC, PSSD, BISI) - Preliminary versions of the OTSC and PSSD software have been successfully run. We are awaiting delivery of a final version of software that will include BISI and EOF terminal software. This software package is expected by October 1, 1983. Pending operational tests and verification of this software package, no firm date for fully operational status of these displays

can be made at this time, however, we are confident that barring unforeseen problems, the displays can be made fully operational by September 1, 1984. We also expect that most displays will be verified and made available well before that date.

4) RDDS - Radiation Data Display System - All equipment, with the exception of a hardcopy display printer, is on site. Installation has been delayed by safety review of the proposed installation's impact on the Plant Radiation Monitoring System. Assuming that the system and its software functions as specified, we plan to have it installed and operating by November 1, 1983.

<u>OPERATIONS STAGING AREA (OSA)</u> All work with regard to the OSA has been completed and this facility is fully functional. The OSA is located in the basement shelter area of the office building. A description of the OSA may also be found in the Donald C. Cook Nuclear Plant Emergency Plan, Rev. 1.

<u>METEOROLOGICAL DATA</u> NUREG-0737, Item III.A.2, "Improving Licensee Emergency Preparedness -- Long Term", pg. III.A.2-2 discusses the requirements for a Class A and Class B dose assessment capability. Our response to this Item is contained in two letters, AEP:NRC:0678A, dated June 28, 1982 and AEP:NRC:0678B, dated August 30, 1982. In these letters we gave the status of four areas related to III.A.2.2. These four areas and an updated status are given below:

(1) Supplementary meteorological information: We stated in AEP:NRC:0678A that the Cook Plant has meteorological instruments on one tower on-site, but that there are two sensors for each of the wind speed, direction and temperature measurements, therefore providing redundancy. We stated that the meteorological data can be interrogated at the Plant and remotely. Since we do not have a second meteorological tower, we consider one set of sensors as providing the primary meteorological measurements system and the redundant sensors as providing the backup meteorological measurements system. With respect to whether a supplemental tower or towers are required to determine airflow patterns and spatial variations of atmospheric stability near a large body of water (the so-called "lake breeze effect" discussed in Regulatory Guide 1.23, Revision 1) we stated in AEP:NRC:0678B that we plan to await further clarification and NRC guidance on this matter prior to considering the desirability of system modifications to the Cook Plant. Based on what we have learned at the March 4, 1983 NRC Workshop on Generic Letter 82-33, it is our understanding that the current Meteorological Measurements Program, as described in the FSAR Section 2, meets the requirements of Supplement 1 to NUREG-0737.

(2) Class A dose assessment capability: As we stated in AEP:NRC:0678A, we have two models for calculating offsite radiation doses. One, designated CPM-002, is an in-house computer program developed by AEPSC. This program is currently being updated to make it more flexible to use and to include additional accident scenario information. The second dose assessment capability exists through Pickard, Lowe and Garrick's MIDAS ACRISO program, discussed below. Both of these programs are undergoing upgrading and trial testing to determine the degree of agreement between their outputs for accident calculations and the outputs of other dose assessment methods which may be used in the event of an accident at the Cook Plant.

It is our intent to use one of these as the primary dose assessment program in the EOF and the other as the backup dose assessment program. The final selection of which one will be the primary dose assessment program and which one will be the backup will be made after the upgraded programs are tested and the results reviewed and compared.

We indicated in AEP:NRC:0678B that we were evaluating a proposal from Pickard, Lowe and Garrick to augment our Class A capability by updating the ACRISO program file to contain in-Plant radiation monitor and accident default information. We stated that when we have completed the upgrading of our model, we will provide you with a description of our Class A dose assessment capability. A Purchase Order was issued to Pickard, Lowe and Garrick in November, 1982 for this work and it is currently in progress.

(3) With respect to a Class A model taking into account complex flow patterns as discussed in Regulatory Guide 1.23, we indicated in AEP:NRC:0678A that the ACRISO program and the CPM-002 program are limited in that they do not account for complex flow patterns that may occur during a lake breeze situation. We stated in AEP:NRC:0678B that we had a Pickard, Lowe and Garrick proposal under evaluation to modify the ACRISO program to utilize a real-time site-specific algorithm for use in turbulent internal boundary layer and lake breeze calculations. However, based upon our information that the NRC was conducting a series of tests along Lake Michigan near the Kewaunee Nuclear Plant to study the "lake breeze" effect, we were reluctant to proceed with a costly revision to the MIDAS program. We therefore stated in AEP:NRC:0678B that we would await further clarification and guidance on this matter prior to making any changes.

Supplement 1 to NUREG-0737 (pg. 13) states that "no changes in existing meteorological monitoring systems are necessary if they have historically provided reliable indication of these variables that are representative of meteorological conditions in the vicinity (up to about 10 miles) of the Plant site. Information on meteorological conditions for the region in which the site is located shall be

available via communication with the National Weather Service. These requirements supersede the clarification of NUREG-0737, Item III.A.2.2". We have not seen the Kewaunee test data, nor have we experienced anything at Cook Plant that shows that the existing arrangement of our meteorological system is inadequate. We therefore believe that the use of our on-site meteorological tower, along with our Class A model, meets the Supplement 1 to NUREG-0737 requirements for dose assessment capability.

(4) Class B dose assessment capability: We indicated that our discussions with two meteorological consultants showed that they do not have a Class B model that fully meets the NRC requirements as defined in NUREG-0654, Revision 1. However, an enhanced Class A model is available which considers site-specific phenomena that affect plume transport. We stated that we expected that operational implementation of this dose assessment capability at the Cook Plant would take place by the NUREG-0737 date of June, 1983 and prior to this date, we would provide the NRC with a description of our plans for a Class B dose assessment capability.

At the March 4, 1983 NRC Workshop on Generic Letter 82-33, in response to a question, the NRC representative stated that Supplement 1 to NUREG-0737 supersedes NUREG-0737, Item III.A.2 "meteorology". Supplement 1 to NUREG-0737 requires meteorological variables, plus a diffusion model that converts these variables into an atmospheric transport model, but that the model need not be a Class A or Class B model as previously characterized in NRC documents. Rather, the model could be an alternative type that models atmospheric transport of radionuclides for the site. The model has to allow real-time dose assessments to be performed, but an elaborate Class B model is no longer required. We are therefore not proceeding with the development of a Class B model for the Cook Plant.

In addition to the direct link from our on-site meteorological tower to the MIDAS ACRISO program, we could manually input to the ACRISO program or to the CPM-002 program meteorological information which could be obtained by voice communications from offsite sources such as the Control Tower at the Ross Field Airport in Benton Harbor, Michigan (12 miles north of Cook Plant) or from regional weather reporting stations in the vicinity of the Plant. Should radiation dose assessments be required out to significant distances from the Plant, or should our protective action recommendations involve the sheltering or evacuation of the public, and there existed reason to believe that the Meteorological Tower on site did not adequately portray the meteorological conditions off-site, we would then use the meteorological conditions obtained via voice communications to the weather service stations and manually adjust the ACRISO program or the CPM-002 program to estimate the radiation doses offsite.

ATTACHMENT 2 TO AEP:NRC:0773
PLAN FOR PHASED IMPLEMENTATION AND INTEGRATION
OF THE EMERGENCY RESPONSE ACTIVITIES

SUBMITTAL OF DCRDR PROGRAM PLAN REPORT (PPR): As indicated in Attachment 1, AEPSC and Torrey Pines Technology, Inc. will prepare the Program Plan Report (PPR) which will be submitted to the NRC on or before September 1, 1983.

The PPR will include, but not necessarily be limited to, descriptions of the following:

1) Staffing and Qualifications of DCRDR Program Personnel.
2) DCRDR Documentation and Document Control.
3) DCRDR Methodology.
4) Human Engineering Discrepancy (HED) Assessment Methodology.
5) Integration methodology of DCRDR with other emergency response activities.
6) A milestone chart of the DCRDR.

It is our understanding, based on the discussions held at the March 4, 1983 NRC Region III Workshop, that activities may begin on tasks within the DCRDR program prior to NRC approval of the PPR.

SUBMITTAL OF DCRDR PROGRAM SUMMARY REPORT (PSR): In acknowledgement of Supplement 1 to NUREG-0737, Item 5.2.b, we recognize the requirement for all licensees to submit a Program Summary Report (PSR) for the DCRDR Program.

We believe, however, that it would be unrealistic at this time, as justified below, to commit to a firm submittal date for the PSR. However, we will submit to you approximately one year following our PPR submittal, the next reasonable and achievable milestone after the specific activities are actually completed.

Supplement 1 to NUREG-0737 specifically requires that the summary report of the "completed review" include (1) "proposed control room changes", (2) "proposed schedules for implementation", and (3) "justification for human engineering discrepancies with safety significance to be left uncorrected or partially corrected". It is not possible to provide in this response the date for submittal of the PSR since much of the details required by Supplement 1 to NUREG-0737 needed to project a submittal date, are not available at this time. For example, the nature and magnitude of potential HED's will not be known until Processes 1, 4, 5 and 6, as defined in the NUREG-0700 guidelines, have been conducted in Phase II of our DCRDR Program. In addition, the time required to complete the Phase III Assessment must also be considered an unknown variable at this time since it is a function of the number of HED's yet to be identified in Phase II. Moreover, it

will not be possible to develop a schedule for implementation of corrective action until Phase III has been concluded.

The date for the intermediate milestone response should be based upon our completion of all the Phase II Review Tasks in our DCRDR Program. For the above justification, we believe it is reasonable to submit to the NRC an "intermediate milestone response" by September 1, 1984, which will include the then current status of our DCRDR Program.

(b) REGULATORY GUIDE 1.97

The present plan is to perform an item by item review for qualification of equipment to the appropriate classification upon implementation of the revised Emergency Operating Procedures.

The Regulatory Guide 1.97, Rev. 2 review will be completed six months following the EOP implementation, together with a replacement schedule being developed thereafter, if one is necessary.

(c) SAFETY PARAMETER DISPLAY SYSTEM

Pending operational tests, no firm date can be made for the operational SPDS system, to the level of availability stated in NUREG's 0578 and 0696. We, however, anticipate that a reasonable date for SPDS functionality would be September 1, 1984, provided the vendor is not required to make unanticipated software revisions.

SPDS INTEGRATION SCHEDULE: It is our intention to make SPDS revisions, found necessary as a result of operational testing, as promptly as possible. This may entail awaiting Plant outages depending on the nature of the revisions. However, it is expected that the SPDS Displays will be sufficiently functional so that the SPDS will be available during DCRDR and EOP preparation. We expect to have the SPDS ready for use during training on the new EOP's.

(d) EOP UPDATE

The Plant Specific Technical Guidelines are under development and are expected to be completed by May 1, 1983.

The Plant Specific Writers Guidelines are under development and are expected to be completed by June 1, 1983.

The Generic Task Analysis is being developed by the Westinghouse Owners' Group and is expected to be completed by August 31, 1983.

The Generic Technical Guidelines have been submitted to the NRC.

The procedures generation package will be submitted to the NRC at approximately half way through the procedures generation process, but no later than three months before formal operator training begins.

The upgraded EOP's will be implemented upon completion of the operator training on the EOP's. This training is presently expected to require approximately one year after completion of the procedures, and is determined by the intentions to include simulator training on the upgrade EOP's for each operator.

The training plan incorporating training on the 82-33 items is essentially completed.

ATTACHMENT 3 TO AEP:NRC:0773

SUMMARY OF EXPECTED COMPLETION DATES FOR EMERGENCY RESPONSE ACTIVITIES

I. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

A.	Delivery of Final Verified Software	October 1, 1983
B.	Begin Operator Training	When Software is Delivered
C.	SPDS Functionality	September 1, 1984
D.	SPDS Available	During DCRDR and EOP Preparation
E.	SPDS Ready for Use	During Training on New EOP's

II. DETAILED CONTROL ROOM DESIGN REVIEW (DCRDR)

A.	Program Plan Report - Submit to NRC	September 1, 1983
B.	Intermediate Milestone Response to NRC	September 1, 1984

III. REGULATORY GUIDE 1.97

A.	Complete the Review of Revision 2 for Equipment Qualification	Six Months after EOP Implementation

IV. UPGRADE EMERGENCY OPERATING PROCEDURES

A.	Complete Plant Specific Technical Guidelines	May 1, 1983
B.	Complete Plant Specific Writers Guidelines	June 1, 1983
C.	Complete Generic Task Analysis by Westinghouse Owners' Group	August 31, 1983
D.	Submit Procedures Generation Package to NRC	Half-way through Procedures Generation Process, but not later than three months before formal Operator Training.

E.	Implement Upgraded EOPs	At completion of Operator Training
F.	Complete Operator Training	One Year after Completion of Procedures

V. EMERGENCY OPERATING FACILITIES (EOF)

A.	EOF - Radiation Data Display System - Installed and Operational	November 1, 1983
B.	TSC-HVAC - Redundant - Fully Operational	September 1, 1983
C.	TSC-RDDS - Installed and Operational	November 1, 1983
D.	Submit Class A Dose Model Description	September 1, 1983