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ALEXICH,M.P.	Indiana & Michigan Electric Co.

RECIP.NAME	RECIPIENT AFFILIATION
EISENHUT,D.G.	Division of Licensing

SUBJECT: Forwards response to Generic Ltr 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events." Vendor document control sys established & program will be in place by end of 1985.

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	NRR/DSI/METB	1	1	NRR/DSI/RAB	1	1
	REG FILE 04	1	1	RGN3	1	1
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INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

November 4, 1983
AEP:NRC:0838A

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 83-28
REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF
SALEM ATWS EVENTS

Mr. Darrel G. Eisenhut, Director
Division of Licensing, NRC
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

This letter is in response to Generic Letter 83-28 of July 8, 1983 which required actions based on generic implications of the Anticipated Transient Without Scram (ATWS) incident at Salem.

Owing to the level of detail associated with these responses, we have included them as an attachment rather than in the body of the letter itself.

Very truly yours,

M. P. Alexich
Vice President

th

Attachment

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
E. R. Swanson, NRC Resident Inspector - Bridgman

A001
1/1

AEP:NRC:0838A

STATE OF OHIO

FRANKLIN COUNTY

Milton P. Alexich, being duly sworn, deposes and says that he is the Vice President of Licensee Indiana & Michigan Electric Company, that he has read the foregoing letter and the Attachment (AEP:NRC:0838A Response to Generic Letter 83-28, Required Actions Based on Generic Implications of Salem ATWS Events) and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

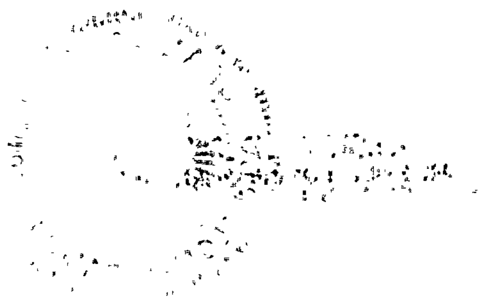
Milton P. Alexich

Subscribed and sworn to before me this fourth day of November, 1983.

Robert A. Barlow

Notary Public

ROBERT A. BARLOW
NOTARY PUBLIC, FRANKLIN COUNTY, OHIO
MY COMMISSION EXPIRES AUGUST '86



ATTACHMENT TO

AEP:NRC:0838A

RESPONSE TO

Generic Letter 83-28

REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

1. POST TRIP REVIEW

1.1 Program Description and Procedure

Presently, at the D.C. Cook Plant, administrative procedures require all shift supervisors to notify the Plant Management of a reactor trip or of any significant operational events. Plant startup is covered by Operating Procedures which require that the Plant Manager's concurrence be obtained prior to restart. In his absence, this responsibility is assumed by the Assistant Plant Manager (operations) or the Assistant Plant Manager (maintenance).

The basis for determining that a restart is acceptable is that the cause of the unplanned trip has been determined and corrected and that the plant reacted "normally."

In the event that the cause of the trip cannot be determined, the Shift Supervisor must determine that reactor operation can proceed safely (i.e., the protective and trip systems are fully functional). Plant Manager or designee approval is required for all reactor restarts.

We are presently developing a reactor post-trip review procedure which will be used to assess the readiness of the unit to safely return to power. This procedure will embody many of the recommendations of INPO's Good Practice, OP-211, Post Trip Review procedure. Extensive use of Sign Off Sheets and plant status logging is being built in so that the plant history around the time of trip can be determined for future analysis, trending, and backtracking.

A first draft of the new procedure has been reviewed by the Plant and AEP engineering personnel. Comments are presently being evaluated.

It is estimated that the second draft will be available for review by January 1, 1984, and the approved procedure issued by March 30, 1984.

1.1.1. Restart Criteria

Present restart criteria are covered above. The post-trip review procedure will include criteria for making the decision to restart.

1.1.2. Responsibility/Authority

The personnel involved with the review and analysis of the events leading to the trip and restart are the Operator, Shift Technical Advisor, Shift Supervisor, Superintendent of Operations, Assistant Plant Managers, and the Plant Manager.

1.1.3. Qualifications/Training

All positions referenced in the procedure meet the qualification requirements of Technical Specification, Section 6.3, Facility Staff Qualifications. The Shift Supervisors, Operations Superintendent, and the Assistant Plant Manager (Operations) hold SRO licenses.

Detailed qualifications of all licensed incumbents have been previously transmitted to the NRC. We can provide these details if requested.

1.1.4. Sources of Information

Item 1.2 (Post Trip Review - Data and Information Capability) provides details of the information available to operating personnel. The new Post Trip Review Procedure will formalize the practice of recording data relating to the trip. It is anticipated that, once finalized, the post-trip review data package will be retained for future reference.

1.1.5. Event Analysis

There is no formal or written criteria for comparing the "trip as occurred" versus the "trip as would have been predicted" in a quantitative sense. However, the Plant has information detailed in Section 1.2 to rely on in this matter.

1.1.6. Independent Review

The Plant Manager has the freedom to ask for an independent assessment of any unplanned trip. In such cases, he is able to call for the advice and assistance of the American Electric Power Service Corporation. The first person he would normally contact for such assistance is the Vice President, Nuclear Engineering, although he would be free to contact any member of the Nuclear Safety Design Review Committee or any member of the Service Corporation that he thought appropriate to solve the problem at hand. If contact with outside vendors was required, this would normally be done after appropriate consultation between the Plant Manager and AEPSC. There is no formal written criteria for taking such an action.

The Plant Nuclear Safety Review Committee will review all events prior to restart if the cause of the reactor trip is not positively known or if the plant response demonstrated an abnormal behavior that has not been corrected. The existing restart procedure ensures the recording of basic information. The new Post Trip Review Procedure will extend the scope and provide more detailed guidance as to what data needs to be saved.

1.2 Data and Information Capability

The following writeup identifies the monitoring systems installed in Unit 1. Identical monitors are installed in Unit 2 and monitor the same devices except for the differences between the Unit 1 and Unit 2 turbines.

A. Hathaway Operations Sequence Monitor.

1.2.1.1. The sequence monitor provides a printed record of the operation of certain selected events. It has the capacity to monitor 192 on-off points and produces a line item output on a printer located in the control room when any one of the monitored points indicates an abnormal condition.

1.2.1.2. Forty-five points are used to monitor events related to reactor trip initiation or reactor trip circuit breaker position, 6 points monitor condensate or hotwell pumps, 13 points monitor feedwater heater extreme high level events, 6 points monitor onsite power diesel generators, 22 points monitor the main feedwater pumps, 40 points monitor the main turbine-generator, and 8 points monitor the step-up and auxiliary transformers and miscellaneous items.

1.2.1.3 The operations sequence monitor permits discrimination for contact closures which occur more than 2 milliseconds apart.

1.2.1.4. A contact closure will result in a line item printout on a dedicated printer located in the control room. The line item contains a 3 digit number for the day of the year, a 4 digit number for the hour and minute, a 2 digit number for the second, a 3 digit number for the milliseconds, an "A" indicating an off normal condition, a 3 digit number to identify the individual input point, and a preprogrammed legend of approximately 22 characters maximum to identify the event to the operator.

1.2.1.5. The printer output sheets may be retained for future use.

1.2.1.6. The operations sequence monitor is powered from a balance of plant (non class IE) inverter.

B. Esterline Angus Turbine Event Monitor.

1.2.1.1. The turbine event monitor is a dual-unit strip chart recorder. Each of the 2 charts has 20 on-off points. The speed of the continuously moving charts is changed after a trip initiation so that 24 hours of chart are advanced through the recorder in 24 seconds. The chart speed then returns to normal until all initiating events have returned to normal and a trip initiation event recurs.

1.2.1.2. Two points, one on each chart, are used to indicate operation of any trip initiation event. These points are used to synchronize the 2 charts. Two points are used to monitor the

Train A and Train B reactor trip circuit breakers, 2 points monitor electrical lockout relays which indicate an electrical system level trip, and 16 points monitor the position of turbine emergency and pre-emergency valves (stop and interceptor valves). The remaining points monitor various turbine trip initiating events.

1.2.1.3. The time discrimination between events is approximately 20 milliseconds when the chart is in high speed operation.

1.2.1.4. The data is displayed on 2 strip charts. Each point operates a heat pen which leaves a continuous trace on the thermally sensitized chart. The pens trace a printed line on the chart to indicate a normal condition. The pen moves off the printed line to a position approximately midway between the printed lines for 2 adjacent points to indicate an off normal condition.

1.2.1.5. The charts form a permanent record and may be retained for future reference.

1.2.1.6. The solenoids which operate the point positioning mechanisms are powered from one of the station batteries. The chart drive mechanism is powered from the balance of plant (non class IE) inverter.

C. Hathaway Oscillograph.

1.2.1.1 The unit oscillograph has 32 galvanometers. Each galvanometer will record one analog channel or, if properly modified, 4 on-off functions. Eight galvanometers have been converted to on-off functions and the remaining galvanometers are reserved for electrical analog quantities. The unit has a prefault recording feature where all input quantities are continuously recorded on a magnetic disc. Under normal conditions, the data are erased and current recordings written over the old space after approximately 100 milliseconds. If one of a specific set of events occurs, the data are recorded on ultra-violet sensitive photographic paper such that the information recorded prior to the event is recorded, followed by additional data resulting from the event. The recording is continued for a fixed time period following the event. Recording chart speed may be selected to be either 12" or 3" per second, the usual practice being to record the initial portion of the event at the higher chart speed followed by additional recording at the slower chart speed.

1.2.1.2. Six points are used to monitor the A and B train reactor trip circuit breaker positions, undervoltage trip initiation, and safety injection actuation; 2 points monitor the start of onsite power diesel generators; 1 point monitors the trip of

the feedwater pumps; 10 points monitor turbine-initiated events; 7 points monitor generator and excitation events; and 4 traces are used for references to assist in identification of trace locations. The analog traces record generator phase currents, phase and ground voltages, and field current.

1.2.1.3. The time discrimination between events during higher chart speed is better than 5 milliseconds between events and better than 10 milliseconds during slower chart speed.

1.2.1.4. The display provided by the developed photographic paper is a reproduction of the amplitude and wave shapes of the analog electrical quantities. The on-off events are indicated by a continuous straight line trace for a normal condition or the absence of the trace at that location, signifying an off normal event. The photographic paper is developed by exposure to ultra-violet light (fluorescent lights are adequate sources), and no wet chemical processes are required.

1.2.1.5. The photographic paper record may be retained for future reference provided it is not continuously exposed to sunlight or fluorescent lights.

1.2.1.6. The solenoids for the on-off event devices are powered from one of the station batteries. The remaining power necessary to operate the oscillograph and accessory equipment is the balance of plant (non class IE) inverter.

1.2.2 The complement of programs for the P250 process computer includes two programs which are relevant to the analysis of reactor trips. These two programs are the Post Trip Review Program and the Sequence of Events Recording Program.

The Sequence of Events Recording Program records the sequence of operation of a number of monitored contacts to a high time resolution. When one of the monitored contacts changes state, an interrupt is initiated which causes the P250 to scan each monitored contact for any change from its previous state. The program stores such changes and the cycle count since the first event. A cycle is nominally 16.7 milliseconds in length. Due to a dead time of 2 milliseconds in the interrupt process, an automatic rebid of the program is programmed for the cycle following each interrupt bid. This is done to avoid loss of contact changes during the dead time. The Sequence of Events Recording Program is terminated when either the cycle count reaches 3600 or 25 contact changes have been recorded.

When the program is terminated, an output routine is called. All collected data are first moved to the output program buffers to free the Sequence of Events Recording Program buffers for continued monitoring. The output routine prints the time of the first event in hours, minutes, and seconds. Following this message, the alpha-numeric address, a 36-character contact description, and cycle count from the first event are printed for each contact change. The first event will always have a cycle count of zero.

The P250 address list indicates that there is an input to the Sequence of Events Recording Program for each potential reactor trip. In the case of reactor coolant pump underfrequency, partial trips are also included. In addition, the reactor trip and reactor trip bypass circuit breakers, main generator output circuit breakers, and turbine stop valves are monitored.

The time discrimination between events is one cycle or nominally 2.0 milliseconds. The format for data display is discussed above in the description of the program. The data is output on one of the P250 typewriters. The printer output sheets may be retained for future reference.

The primary power source for the P250 computer is an inverter supplied by the AB battery and 600 volt bus 11B. If the inverter should fail, the P250 computer would be switched by an automatic bus transfer to the control room power distribution circuit, CRP-3, which is supplied from the plant lighting transformer. The power source is balance of plant (non class IE).

The Post Trip Review program periodically records a number of pre-selected inputs. These inputs are stored on a disc in a circular buffer, with newer sets of data replacing the older sets. When a trip occurs, either automatically (Post Trip) or manually (Test Trip), the pre-trip data being entered into the circular buffer are frozen and the data are thereafter stored in a post-trip buffer. When this buffer is filled, both sets of data (pre and post) are printed out on the typewriter.

The parameters monitored are analog in nature. They would normally include steam generator feedwater flow and steam flow, steam generator narrow range and wide range level, pressurizer level, pressurizer level setpoint, source range detector output, intermediate range detector output, power range detector output, first stage turbine pressure, steam generator pressure, pressurizer pressure, containment pressure, unit gross electrical output, Taverage, delta T power, overtemperature delta T setpoint, overpower delta T setpoint, wide range cold leg temperatures, pressurizer steam temperatures, T-reference, auctioneered delta T, and auctioneered Tavg. These parameters remain as selected by the computer vendor, Westinghouse Electric Corporation.

Eight of the parameters in the previous paragraph are sampled at 2-second intervals, 6 seconds before and after the trip. These are total NIS power range power, turbine first stage pressure, unit gross electrical output, and auctioneered Taverage. All parameters are sampled at 8-second intervals for 2 minutes before and 3 minutes after the trip. These sampling times remain as set by the computer vendor.

The Post Trip Print program first outputs a start message on the appropriate typewriter. It then outputs a line of headings for the values which will be printed in columnar form. The headings consist of the six-character name of the point. The values are printed below their names, starting with the oldest set of data on the first line, the next oldest on the next line, and so on until the most recent pre-trip data are printed. Included in each row of data is a column indicating the time. When all the pre-trip data for this set of points are printed, the message POST TRIP DATA - TRIP TIME XXXX is printed. All the post-trip for these points are printed in the same format as described above.

After all the post-trip data for these points are finished, the program starts over with another set of data in the same format: 6 character names, pre-trip values, trip message, and post-trip values. When all the points have been printed, the program outputs a finished message, unblocks the collection programs, and exits.

From time to time other analog values may be substituted for the identified quantities to implement investigations of anomalous behavior of other systems or components.

The discussion of data retention and power source for the Sequence of Events Recording program also applies to the Post Trip Review program.

1.2.3 In addition to the data collection previously mentioned, the information contained on the following strip chart recorders would assist in resolving or reconstructing the events surrounding a reactor trip.

RCS Wide Range Loop Temperature Hot and Cold Leg Recorder (4)

Reference Tavg Temperature - Auctioneered Tavg Temperature Recorder

Control Bank and Insertion Limits Recorder

Nuclear Instrumentation Recorder (7)

Main Steam, Main Feedwater Flow and Steam Generator Level Recorder (4)

Feedpump Suction and Discharge Pressure Recorder

Main Turbine Vibration Recorder

Feed Pump Turbine Vibration Recorder

Main Steam Pressure Recorder (2)

First Stage Pressure Recorder (2)

Pressurizer Level Recorder

Pressurizer Pressure Recorder

Computer Trending Recorder (2)

Wide Range Reactor Coolant Pressure Recorder

Containment Pressure Recorder (2)

The preceding is not a listing of all recorders available, but is a listing of those relevant to a unit trip.

2. EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE

2.1 Reactor Trip System Components

For the reactor trip system, a list of components is enclosed (Attachment 1). Included in this list are the sensors which measure the trip variables and the circuits up to and including the reactor trip breakers. The components downstream of the breakers, such as the rods themselves, are excluded.

We do not control safety-related components in the manner indicated in the portion of the position stated in 2.1 of Generic Letter 83-28 that states, ". . . all components whose function is required to trip the reactor are identified as safety related on documents, procedures, and information handling systems used to control safety-related activities including maintenance, work orders, and parts replacement." Rather, our method of control is described below, as is the historical perspective of how it evolved.

At the conceptual design stage, AEPSC issued a series of Nuclear Safeguard Design Memos within the Corporation to convey, among other things, the principles on which safety classifications were to be made.

At the construction stage, the Quality Control Manual for the D.C. Cook Plant (Rev. 3 of 1971) stated: "Those items vital to safe shutdown and isolation of the reactor, or whose failure might cause or increase the severity of a loss of coolant accident, or result in an uncontrolled release of excessive amounts of radioactivity are designated class I. Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected and erected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance."

These documents constituted the original basis for equipment classification which, to some extent, is used today. However, present Corporate Procedures have modified these classifications to define safety-related components as follows:

"Safety Related (N Items) are those:

1. Which are associated with the
   a) safe shutdown (hot) of the reactor, or
   b) isolation of the reactor, or
   c) maintenance of the integrity of the Reactor Coolant System Pressure Boundary.

   OR

2. Whose failure might
   a) cause or increase the severity of a design basis accident as described in the FSAR, or
   b) lead to a release of radioactivity in excess of 10 CFR 100 limits."

Electrical, civil, mechanical, and structural items with known exceptions, are listed as safety related if they are:

1. Seismic class I,
2. Electrical class IE,
3. Associated with the Engineering Safety Features Actuation System,
4. Associated with the Reactor Protection System, or
5. Safety-related for other reasons, such as portions of the Fire Protection System and Post-TMI installations.

Components which met the above requirements (with known exceptions) were classified as safety related (N-Items) and entered into a computerized list called the N-List.

A component is the smallest unit which is separately identified in original equipment procurement documentation. In other words, if a motor control center was bought as a unit, each sub-assembly within that MCC would not be separately entered in the N-list. Structural items such as concrete, rebar, building steel, etc. were covered by a line entry per item. Electrical items such as relays, switches, conduits, fittings, and trays were also covered by a single line entry per item.

The N-list is one of the documents utilized to identify equipment as being safety related. However, it is not all inclusive and therefore not the sole source of reference. When specific determinations must be made, other documents such as the FSAR, Technical Specifications, related communications to the NRC, flow diagrams, isometrics, electrical one line, and elementary diagrams are also consulted.

The control of the N-list, including update and maintenance responsibilities, is set forth in corporate level General Procedures.

If anyone within AEP or the Plant is unsure of the classification of a component, he is required to check with the responsible AEPSC Cognizant Engineer.

Under the present system of work control, a Job Order is prepared for all repair and modification work performed at the plant. Attachment 2 is a sample Job Order. During the Job Order preparation process, the safety classification (N or S) of the equipment, as well as procedures required to perform the work, are indicated on the Job Order form. The personnel responsible for designating the safety classification are experienced and knowledgeable in the use of the reference documents described above. The same reference documents, including purchase orders and the DCC Specifications, are used to assure that qualified replacement parts are utilized when required.

The principal supply of components whose function is required to trip the reactor was through Westinghouse Electric Corporation. We have received from Westinghouse a list of all changes recommended for the equipment they have supplied for the D.C. Cook Nuclear Plant. Verification by formal written documentation has not been obtained on

all these items; however, Attachment 3 lists the modifications and their disposition, including the reason(s) why the change was not incorporated. Attachment 4 describes the controls that apply to changes affecting the D.C. Cook Nuclear Plant. We believe that these attachments demonstrate that our program is effective in accomplishing safety objectives.

To allow us to systematically track vendor documentation for the lifetime of the plant, we are setting up a Vendor Document Control System. Attachment 5 gives a preliminary description. We currently anticipate having a program completed and in place by the end of 1985.

Based on the above, it is our belief that we currently have an effective program to assure that all components whose function is required to trip the reactor are treated in a manner which is not inimical to public health and safety and, although not the same as the NRC stated position, will accomplish what we believe to be the intended safety objective of the NRC.

We have reviewed the trip components list to make sure that, except as noted on Attachment 1, each item is on the N-list and, thus, under the controls cited above. We cannot at this time submit a statement confirming that they conform to the position regarding equipment classification because, as described above, we have a different program in this area.

Development of the program cited in Item 2.1 of Generic Letter 83-28 would constitute a major undertaking and would, in all likelihood, take considerable time. If for any reason, the NRC should find our response in this area to be unacceptable, we would like to meet with your staff to discuss alternative approaches.

2.2 Equipment Classification and Vendor Interface (Programs for all Safety-Related Components)

This subject is being reviewed in depth within AEPSC and the D.C. Cook Nuclear Plant.

In view of the considerable scope of the programs involved, the amount of detailed information necessary, and the input from INPO on vendor interface, we anticipate that we will need until March 30, 1984 to respond to this section.

3. POST MAINTENANCE TESTING

3.1 Reactor Trip System Components

3.1.1 Post-Maintenance Testing

We have reviewed our test and maintenance procedures and Technical Specifications and have determined that post-maintenance operability testing of reactor trip system components is required to be conducted prior to returning the system or components to service. The testing demonstrates that the equipment is capable of performing its intended safety function. It should be noted that there are components such as control switches, inverters, batteries, and cable penetrations that are verified operable as part of a system test or by verifying output parameters.

Post-maintenance operability verification requirements for work not covered by a specific procedure is reviewed and established as part of the job order review process, the plant design change procedure, and the clearance permit system. Plant instructions governing maintenance work and design changes are presently undergoing review as part of our Regulatory Performance Improvement Program. This review includes specific instructions to address both post-maintenance testing and design change installation operability testing.

3.1.2. Recommended Tests

We have received from Westinghouse their Technical Bulletin NSD-TB-83-05 which lists all their Technical Bulletins and Data Letters sent to members of the Westinghouse Owner's Group as of June 20, 1983. We have reviewed all these bulletins and for those that pertain to the trip system, we have assured ourselves that the information was conveyed to cognizant engineers at AEPSC and the relevant Department at the Plant. Changes in testing and maintenance procedures prompted by these bulletins have been initiated.

For vendors other than Westinghouse, we have started to send out letters requesting a list of their documents that pertain to reactor trip components so that we may verify their incorporation. Conditional to our receiving the information from our vendors on a timely basis, we will attempt to verify the disposition of all of these modification notices prior to March 30, 1984.

3.1.3. Excessive Technical Specification Mandated Testing

We are aware of the NRC Task Force on Plant Technical Specifications Surveillance Requirements and its efforts to enhance safety by seeking out optimum post-maintenance testing frequencies. AEPSC is a member of the Westinghouse Owner's Group (WOG) which has an active subcommittee on Technical Specifications.

On September 16, Mr. J. J. Sheppard, the chairman of WOG, wrote to Mr. J. Sniezek, Director of the DEDROGR staff, outlining the Owner's Group position on this issue. We support the position and philosophy contained in the letter, particularly the comment on too frequent startup testing of diesels. We do not intend at this time to separately submit an application for reduced testing with supporting justification to demonstrate that safety is degraded rather than enhanced.

3.2 <u>All other Safety-Related Components</u>

Due to the large volume of activity associated with this response, we are currently scheduling our response in this matter to be issued on March 30, 1984 for items 3.2.1 and 3.2.2. For item 3.2.3, our response is the same as that presented under item 3.1.3.

4. REACTOR TRIP SYSTEM RELIABILITY IMPROVEMENTS

4.1 Vendor-Related Modifications

Appendix A of Westinghouse Technical Bulletin NSD-TB-83-05, dated June 20, 1983, has been reviewed by AEPSC Cognizant Engineers. Except for those bulletins listed in Attachment 6, all those changes that relate to our plant have been or are being implemented. Plant personnel are aware of the contents of bulletins listed in Attachment 6. A review of the Plant procedures is underway to ensure that necessary updates are made.

The modifications recommended in NCD-Elec-18 relate to certain problems identified in DB-50 reactor trip breakers prior to delivery to the D.C. Cook Plant. Corrections were made by the manufacturer. The Westinghouse Owner's Group sent us a letter (83-242 of September 15, 1983) asking for visual confirmation that we had a post-1972 UVTA Unit. This was done at the Cook Plant. Notice NCD-Elec-18 does not apply to our plant.

We have verbally been informed by Westinghouse that additional Bulletins (such as 83-02 Rev. 1) will be sent to us in the very near future. When we receive them we will act on them.

4.2 Preventative Maintenance and Surveillance Program for Reactor Trip Breakers

4.2.1 Periodic Maintenance

The current maintenance program (except for frequency) for the reactor trip breakers is in compliance with the Westinghouse recommendations contained in Westinghouse Technical Bulletin 83-02. Prior to the issuance of this bulletin, the procedure was in compliance with the recommendations of Bulletin 74-02. The circuit breakers are inspected, cleaned, and lubricated during each refueling outage. Critical clearances are verified to be within tolerance by use of go/no-go gauges. Instructions for adjustments are included in the maintenance procedure.

4.2.2 Parameter Trending

The performance of the undervoltage trip device will be monitored during scheduled inspections by reducing the applied voltage to the undervoltage trip coil and recording the trip voltage. At least three values for each circuit breaker will be obtained. Future values will be compared for individual circuit breakers to monitor any changes. Comparisons will also be made between circuit breakers. Criteria for evaluating changes are being developed and are expected to be in place by the end of 1984.

4.2.3 and 4.2.4 **Life Cycle Testing and Periodic Replacement**

Life cycle testing of the shunt trip attachment (STA) and the undervoltage trip attachment (UVTA) of the reactor trip switchgear is being conducted by Westinghouse for the Westinghouse Owner's Group. This program is aimed toward establishing the service life of these devices and substantiating periodic test requirements with proper maintenance.

The UVTA testing for the DB-50 breakers is complete and we are expecting a report by the end of November 1983. The balance of the program is scheduled for completion in the second quarter of 1984.

Any changes to preventative and corrective maintenance or replacement intervals indicated by the results of the tests will be considered for incorporation into our existing procedures.

4.3. <u>Automatic Actuation of the Shunt Trip Attachment for Westinghouse and B&W Plants</u>

We intend to modify the reactor trip system so that in case of an automatic trip, both the Shunt Trip Attachment (STA) and Under Voltage Trip Attachment (UVTA) will cause the reactor trip breakers to open.

The Westinghouse Owner's Group asked Westinghouse to design this automatic actuation feature. A detailed generic design package was developed and transmitted to Mr. Eisenhut by J. J. Sheppard (Chairman, WOG) by letter on June 14, 1983.

The NRC issued a Safety Evaluation Report on the generic design on August 10, 1983 (letter D. Eisenhut to J. J. Sheppard). Several questions were raised and additional information requested in that Report. The Report concluded that the generic design was acceptable but pointed out the plant-specific information required. We will consider the STA to be a safety-related function and stipulate that the circuitry used to implement it must be class IE.

Westinghouse is proceeding to qualify the STA to IEE 323-74 and 344-75 criteria and to conduct seismic tests and issue a complete test report. The present schedule calls for completion of their work in seven months. On receiving a final generic design from Westinghouse, we will be starting a plant-specific design. A design package will be submitted to the NRC for safety evaluation. We are unable to give you an implementation schedule before receipt of the complete Westinghouse design package and report.

4.4. Improvements in Maintenance and Test Procedures for B&W Plants

This Section does not apply to the D. C. Cook Plant.

4.5 System Functional Testing

4.5.1. STA and UVTA Testing

The reactor trip breakers are currently tested on-line by operation of the undervoltage trip device.

The present arrangement of four circuit breakers, two trip and two bypass, permits on-line testing of the breakers. We currently test the UVTA prior to every start-up and once every three months during unit operation. The shunt trip is independently tested prior to each start-up. No failures have been encountered.

Once the modification to enable automatic actuation of the shunt trip attachment is made, we will test both the UVTA and the STA while the unit is in operation.

4.5.2. On-line Testing

Since we perform on-line testing, this section does not apply to our plant.

4.5.3. Frequency of On-line Testing

Our position is as follows:

The reactor trip circuit breakers at the D. C. Cook Plant are installed in a clean and dry location and are not subject to any deleterious environmental influences. The present maintenance program requires that the circuit breakers be serviced at every refueling outage. At this time the mechanical features of the circuit breakers are inspected and adjusted as necessary to maintain the critical clearances determined by the manufacturer to be necessary for reliable operation. The circuit breaker and its compartment are cleaned and lubrication is applied as recommended by the manufacturer. The main contact resistance is verified to be acceptable by test. The circuit breakers are then installed in the metal clad enclosures.

Prior to returning the circuit breaker to service, an electrical functional test is performed which tests the electrical closing, electrical shunt trip and the undervoltage trip. In compliance with the Technical Specifications, the undervoltage trip of each circuit breaker has been tested on line at three-month intervals.

There have been no failures of the reactor trip circuit breakers to trip during tests or in actual operation. This history of excellent performance has been maintained for over 7 years for one unit and 4 years for the other. The present

surveillance and maintenance program has been adequate, resulting in no failures.

The present maintenance schedule permits inspections and adjustments to the circuit breakers at intervals which are more frequent than necessary, considering the clean environment and light electrical service required of the circuit breakers. The testing at three-month intervals has provided the necessary exercise to ensure freedom of motion of the circuit breakers and its attachments when they are called on to operate to perform their safety function. Assuming a maximum test interval of one hour for each on-line surveillance test for each circuit breaker, the reactor protection system is dependent on one safety train for tripping for two hours every three months. Increased on-line testing frequency will result in greater time dependency on one train for tripping without increasing the assurance that the reliability of the circuit breakers has been improved.

The Westinghouse Owner's Group is carrying out tests on these breakers. We expect to receive their results in February 1984. If there are any changes we feel we need to make as a result of the tests, we will communicate to you by March 30, 1984.

ATTACHMENT 1 to AEP:NRC:0838A
REACTOR TRIP SYSTEM COMPONENTS

INSTRUMENTS

Steam Generator Level

BLP - 110
120
130
140
111
121
131
141
112
122
132
142

Main FeedWater Flow

FFC -210
220
230
240
211
221
231
241

Main Steam Flow

MFC - 110
120
130
140
111
121
131
141

Steam Pressure

MPP - 210
220
230
240
211
221
231
241
212
222
232
242

Reactor Coolant System Flow

NFP - 210
220
230
240
211
221
231
241
212
222
232
242

Pressurizer Level

NLP - 151
152
153

Pressurizer Pressure

NPP - 151
152
153
NPS - 153

Reactor Coolant Temperature

NTP - 110
120
130
140
111
121
131
141
210
220
230
240
211
221
231
241

Nuclear Flux Monitors

N - 31
32
35
36
41
42
43
44

Containment Pressure

PPP - 300
301
302

Turbine 1st Stage Pressure

MPC - 253*
254*

Turbine Control Fluid Pressure

LPS - 90*
91*
92*

CONTROL SWITCHES

101-RTA
101-RTB
6-SIA1
6-SIA2
6-SIB1
6-SIB2
5-PSIA
5-PSIB
5-SSIA.
5-SSIB

RELAYS

96-DUF
96-1AUF
96-1BUF
96-1CUF
96-1DUF
96X-AUFB
96X-BUFB
96X-AUFA
96X-BUFA
96X-CUFB
96X-DUFB
96X-CUFA
96X-CUFB

ATTACHMENT 1 (Cont'd)

RELAYS	MISCELLANEOUS	
27-AUV	4KV BUS 1A P.T.	*
27-BUV	4KV BUS 1B P.T.	*
27-CUV	4KV BUS 1C P.T.	*
27-DUV	4KV BUS 1D P.T.	*
27-1AUV	ACB 52-1B9 AUX. SW.	*
27-1BUV	ACB 52-1C2 AUX. SW.	*
27-1CUV	ACB 52-1D9 AUX. SW.	*
27-DUV	ACB 52-1A4 AUX. SW.	*
62-27X-AUVB	33-SVS-1	*
62-27X-BUVB	33-SVS-2	*
62-27X-AUVA	33-SVS-3	*
62-27X-BUVA	33-SVS-4	*
62-27X-CUVB		
62-27X-DUVB		
62-27X-CUVA		
62-27X-DUVA		
96-AUF		
96-BUF		
96-CUF		

*Non N-grade component

Rationale for the use of non N-grade components in the Reactor Trip System:

Some devices which provide contact closures to the Reactor Protection System are not included on the "N" list. These devices are located in areas which do not meet the requirements for safety-grade structures such as the turbine building or are connected to equipment which has no other safety function such as the reactor coolant pump circuit breakers and their source electrical busses. The trip functions provided by these contacts are used to develop anticipatory trips.

The bus potential transformers used as transducers for the four reactor coolant pump motor source electrical busses are connected to non-safety grade busses. The potential transformers cannot be safety grade or class IE if their enclosure is not also class IE. These potential transformers provide the voltage to the undervoltage and underfrequency relays which provide an anticipatory trip of the reactor on low reactor coolant flow. These bus potential transformers are identified as "4kv BUS 1A PT" for the 1A bus. Similar designations are used for busses 1B, 1C, 1D, 2A, 2B, 2C, and 2D.

The reactor coolant pump circuit breaker position switches are located in circuit breakers in non-safety busses. These circuit breaker position switches provide an input to the logic to develop an anticipatory trip of the reactor on trip of reactor coolant pumps. These switches are identified as "ACB-52-1B9 aux sw." for circuit breaker 1B9. Similar identification is used for circuit breakers 1C2, 1D9, 1A4, 2B9, 2C2, 2D9, and 2A4.

Two sets of devices on the main turbine provide input contacts to the reactor protection logic system. The turbine hydraulic trip system pressure is sensed by pressure switches located in the turbine front standard. A loss of pressure indicates that the turbine has been tripped. These devices are identified as LPS-90, LPS-91, and LPS-92. Two sets of limit switches on each turbine main stop valve also provide input contacts to the reactor protection logic system to indicate closure of the main stop valves. Closure of all four main stop valves is a redundant indication of turbine trip. The trip system pressure switches and the main stop valve limit switches are located on the main turbine which is not a safety-related device. The devices develop anticipatory trips. These switches are identified as "33-SVS-1", "33-SVS-2", "33-SVS-4."

Two pressure transducers, MPC-253 and MPC-254, are located adjacent to the main turbine and are used to sense the turbine first stage pressure as a measure of turbine load. These transducers are not listed on the "N" list because they are turbine-mounted devices.

Exclusion of all of the above items from the N-list does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the public.

ATTACHMENT 1 (Cont'd)

MISCELLANEOUS

Reactor Trip Switchgear
Solid State Protection System - Input, Output, Logic Cabinets
Reactor Protection Channel 1, Cabinets 1-4
" 2, " 5-8
" 3, " 9-11
" 4, " 12-13
Reactor Protection Aux Cabinets Aux Cabinets RPSX-A and RPSX-B
Control Panel - FLX
" - PR2
" - SIS
" - SG
" - SAR
Cable - Item # 3064
" 3073
" 3074
" 3075
" 3076
" 3077
" 3111
" 3112
" 3119
" 3120
" 3121
" 3122
" 3123
" 3124
" 3125
" 0871
" 0872
Electrical Instrument Penetration EP-08
" EP-10
" EP-11
Electrical Instrument and Control Penetration EP-12
Inverter and Distribution Cabinet CRID I
" " CRID II
" " CRID III
" " CRID IV
250V DC Plant Batteries

DONALD C. COOK PLANT ATTACHMENT 2 TO AEP:NRC:0838A

Unit No. | Acct/W.O. No. | File | 14998

COMPLETED BY ORIGINATING DEPARTMENT

Bldg. ______ Elev. ______

Requested By ______ Date ______

Supv. Appv. ______ Dept. Hd. ______

Tech Spec Limited ___ Yes ___ No

Time Limit ______ Reference ______

Time Limit Starts ______

CLASSIFICATION

24 Hour ☐ Regular ☐

Load Reduction ☐

Outage ☐ Hot ☐ Cold

WORK TO BE PERFORMED BY

Maint. ☐ Tech. ☐

Other ☐

SOE/UF Approval to Start ☐ Yes ☐ No

Condition Report Submitted ☐ Yes ☐ No

RFC No. ______

Job Description ______

Dept. Appv. ______ Issued To ______ Coordinate With ______

	Yes	No		Number		Permits		Number
Flow Diagram Ref.						Clearance		
Procedure Required						Weld, Grind, Burn		

COMPLETED BY DEPARTMENT RESPONSIBLE FOR WORK

______ Check With RP Section for RWP Permit Issued ☐ Yes ☐ No Number ______

______ Check With Chem Section for Rework Permit Issued ☐ Yes ☐ No Number ______

(N) System ☐ Yes ☐ No

Other Instructions ______

SOE Approval ______ Service Hour/Operation Counter ______

Description of Work Done ______

Supplemental J.O. Submitted ☐ Yes ☐ No By ______

Retest required ☐ Yes ☐ No Performed or observed by ______

SOE/UF Notified of Work Completion ______

Work Performed By ______

Date Completed ______

Work Area Inspected By ______

Post Job Order Review Performed with Consideration Given to NPRDS, Condition Report, Data History and Adequate Documentation

By ______

Tech Spec Time Limit:

Start Time ______

Date ______

Job Completed, Operability Verified,

Returned to Operation

Time ______

Date ______

ATTACHMENT 3 to AEP:NRC:0838A
LIST AND DISPOSITION OF WESTINGHOUSE RECOMMENDED
CHANGES AFFECTING REACTOR TRIP SYSTEMS

Bulletin	Disposition
75-17 122975.	Loose terminal connections for power supply inverters. This applied to Westinghouse Inverters and does not apply to D.C. Cook plant since solid state inverters are used.
77-10 072177.	AR relays with latch attachment. All relays were inspected and those having the affected date codes were purged from the installed system or from stock. No further action is required at this time.
79-02 041779.	Overcurrent trip devices on DB circuit breakers. Not applicable to the safety systems of D.C. Cook plant. No action required.
82-03-R1 121482.	AR relays with ARMLA attachments. We have the relays and have qualified them seismically. No further action is required at this time.
83-02 032483.	Reactor trip breaker maintenance. Recommended action has been incorporated in plant maintenance procedure for reactor trip circuit breakers.
83-03 032483.	Reactor trip circuit breaker testing. Test procedures have already incorporated separate test for undervoltage and shunt trip devices.

ATTACHMENT 3 (Cont'd)

NSID Letter	Disposition
63-02 061063.	Tutorial information on noise coupling into instrument circuits. No specific action identified. No action required at this time.
68-14 051568.	Tutorial information on lead/lag instrument systems. No specific action identified. No action required at this time.
68-25 073168.	Misuse of vital instrument bus. General information and precautions. No specific action identified. No action required at this time.
68-26 073168.	Misuse of vital instrument bus. General information and precautions. No specific action identified. No action required at this time.
68-26 080768.	Unauthorized field modifications. Applied to turnkey installations. Does not apply to D. C. Cook Plant. No further action required.
70-18 091870.	Amphenol Triax Cable defects. No Amphenol triax cable is used at D. C. Cook Plant. No further action is required.
72-80 080272.	Inverter tests. Does not apply to D. C. Cook Plant since Westinghouse inverters are not used.

ATTACHMENT 3 (Cont'd)

INSTRUMENTATION & CONTROL

Technical Data Letter No.	Disposition
69-4	Units Referenced in letter (M/63S and M/635DR) are not at our facility. Our units are 63R, 63U, 63U-D or 63U-E, F.
70-21	Drop Time Test Procedure 12THP4030STP.387 is in accordance with letter.
71-17SU	Letter impacts initial startup and would not affect operating plant.

Technical Bulletin	Disposition
73-6	The Coolant flow procedures THP6030IMP.099 (U-1) and THP6030IMP.199 (U-2) are properly scaled for the bistable inputs. Secondary steam flow has a square root extractor before the bistable input.
73-19	Connections involve the operating coil and would not hamper a reactor trip.
73-20	Plant presently exercises these pots during calibration.
75-6	Review of Westinghouse-supplied amplifiers indicates none are the suspect models. Ref. AEP/AMP WNES Instrumentation manual indicates all amplifiers are Foxboro Model 66BC-O.
77-11	Plant procedures THP4030STP.045 (U-1) and THP4030STP.145 (U-2) utilize the procedure reference in this Tech Bulletin.
78-2	Would not inhibit a trip.
81-12	No Foxboro 92B's are being used at our plant.

ATTACHMENT 3 (Cont'd)

Data Letter	Disposition
80-03 022980.	Amphenol Instrument connectors. Rexolite insulated connectors. We will initiate a new attempt to obtain connectors with the recommended insulation. Attempts to obtain these connectors from Amphenol were unsuccessful in the past.
81-02 062481.	Instrument inverter transformer redesign. Does not apply to D. C. Cook Plant since Westinghouse inverters are not used.
82-07 041382.	Instrument inverters. Applies to Westinghouse inverters. Not applicable to solid state inverters.

ATTACHMENT 3 (Cont'd)

The remaining items listed below were determined not to impact the Reactor Trip Function:

Technical Bulletins:

73-1	75-5	76-1	76-15	78-1-R1
73-7	75-5 Suppl A	76-3	77-3	78-3
73-15	75-07-R1	76-4	77-4	78-4-R1
73-16	75-8	76-6	77-5	79-3
74-4	75-0	76-7	77-6	79-4
74-8	75-12	76-8	77-7	79-6
74-14	75-13	76-9	77-8	79-7
74-14 Suppl A	75-14-R1	76-11-R1	77-9	79-8
75-1	75-15	76-13	77-12	79-9
75-3	75-18	76-14	77-14	80-1
80-2	81-1	81-0	82-5	
80-3	81-2	81-10	82-6	
80-4	81-3	81-13	82-7	
80-7	81-4-R1	82-1	83-1	
80-10	81-7	82-2	83-4	
80-11	81-8	82-4		

NSID Data Letters

59-3	68-19	69-21-R1	71-4	73-3
61-3	68-27	69-23	71-8SU	73-9
62-4	68-29	69-24	71-9SU	74-1
63-1	68-31	69-29	71-12	74-4
63-4	68-32-R2	70-2	71-14	74-7
63-5	68-33	70-3	71-15	75-2
64-2	68-36	70-6-R3	71-16	75-6
64-4	68-37	70-8	71-20	75-7
67-4	79-3	70-9	71-21-R1	76-2
68-1	69-7	70-12	72-1	76-3
68-2	69-9	70-15	72-3SU	76-5
68-4	69-10	70-16	72-4	77-2-R1
68-5	69-11	70-17	72-5R1	77-4
68-7-R1	69-12-R3	70-20	72-6SU	77-6
68-8-R1	69-14	70-22	72-7	77-7
68-11	69-15-R2	70-25	72-9SU	78-1
68-12	69-16	71-1	72-12SU	78-2
68-14	69-19	71-3SU	73-1SU	78-3
68-15	69-20			
78-4		82-12		
78-5	80-5-R1	82-13		
78-6	80-6			
78-7	80-9			
78-8	80-10			
78-9	81-1			
78-10-R1	82-1			
78-11	82-2			
78-12	82-3			
79-3	82-4-R1			
79-5	82-5			
79-6	82-6			
79-7	82-8			
80-1	82-9			
80-2-R1	82-10			

ATTACHMENT 4 to AEP:NRC:0838A

CONTROL OF DESIGN CHANGES
DONALD C. COOK NUCLEAR PLANT

Changes to safety-related equipment, systems, and/or structures (engineering design changes) are controlled by a Corporate level procedure. This procedure is supplemented by lower tier organizational unit specific procedures if necessary.

The vehicle for initiating, reviewing, approving, designing, and implementing an engineering design is the Request For Change (RFC). RFCs may be initiated either by the Cook Plant staff or AEPSC.

RFCs, regardless of location of origin, are assigned to a Lead Engineer in the cognizant AEPSC engineering or design section. The Lead Engineer is responsible for the development of the proposed change and the coordination of interdisciplinary review, scoping, and costing.

Once the conceptual design for an RFC has been completed, the RFC is reviewed and approved by the Lead Engineer's supervisor, and routed to the AEPSC Nuclear Safety & Licensing Section (NS&L) for review. NS&L reviews the RFC for safety implications, highlights any safety issues and safety-related features, and assures that the proposed change will not create a significant hazard to the health and safety of the public and does not constitute an unreviewed safety question as defined in 10 CFR 50.59.

Upon completion of NS&L's review, the RFC is forwarded to the Cook Plant, Plant Nuclear Safety Review Committee (PNSRC). The PNSRC reviews the RFC for safety implications, desirability, necessity, and also considers the operability and maintainability of the change (from a safety standpoint) during both installation and subsequent service.

Once the PNSRC accepts the RFC, it is routed to the Plant Manager for his review and acceptance. The Plant Manager then routes the RFC back to AEPSC for submittal to the Change Control Board (CCB).

The CCB reviews the RFC to assure that proper consideration has been given to the impact that the proposed change may have on plant safety, test programs, procedures, costs, schedules, and current plant status. The CCB determines that the proposed change gives proper consideration to good engineering, design, construction, and operating practices for nuclear plants, and that the proposed change meets the requirements of applicable codes, standards, and guidelines.

After CCB approval, a copy of the RFC is forwarded to the AEPSC Nuclear Safety Design Review Committee for off-line independent review. The NSDRC is mandated by the Cook Plant Technical Specifications to independently review the safety evaluation performed by NS&L for changes to equipment or systems completed under the provision of 10 CFR 50.59 to verify that the change did not constitute an unreviewed safety question.

The original RFC, upon CCB approval, is returned to the Lead Engineer for initiation of the "interactive" engineering design process. This results in the generation of the detailed design required to implement the change. In parallel with this process, engineering documents impacted by the change are revised.

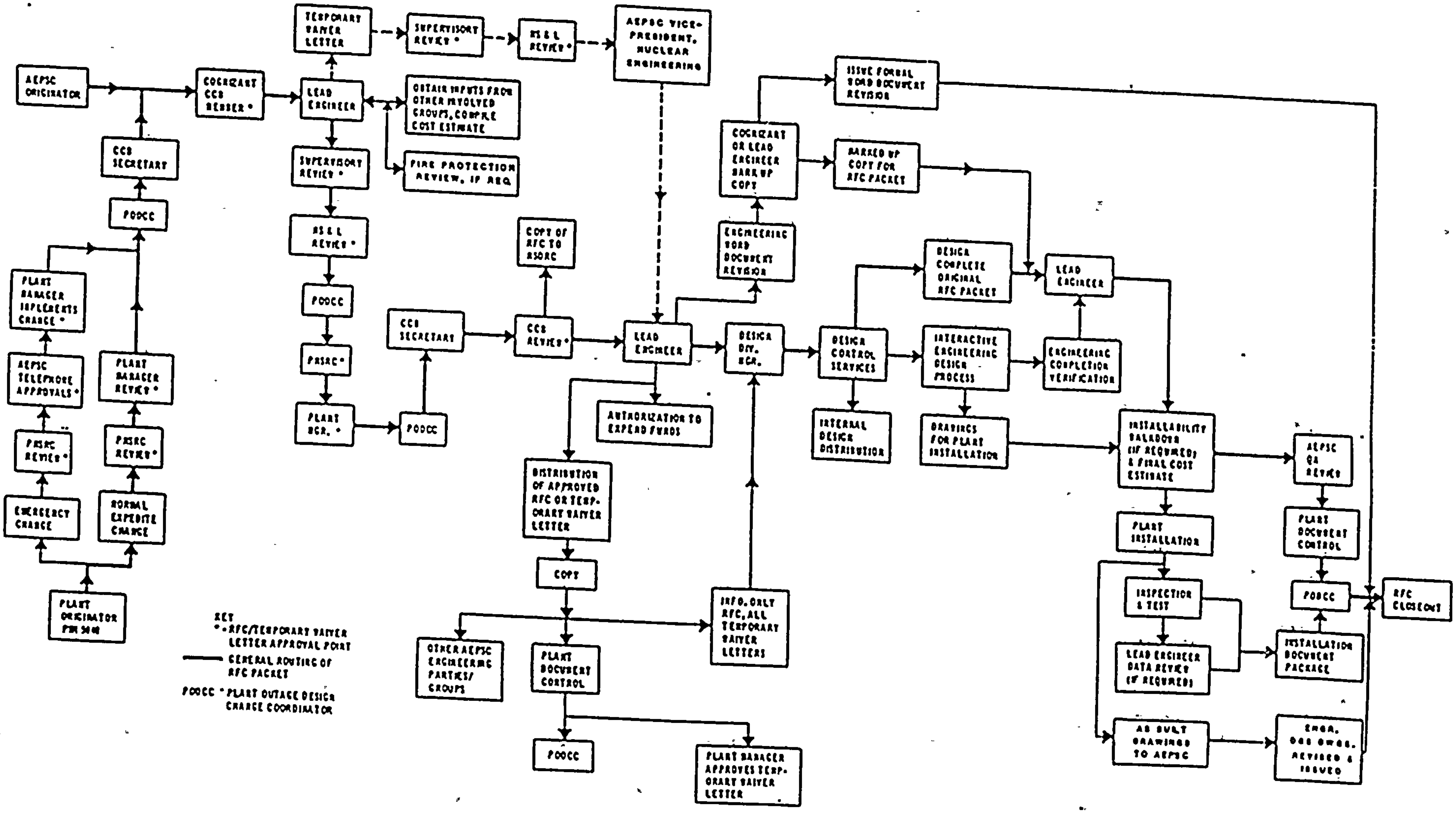
This activity is completed when all design documents necessary to implement the change have been issued. Prior to onsite implementation of a change, a checklist (RFC Installation Authorization) must be completed and submitted for approval by plant management. This list identifies:

- any precautions and limitations,
- the individual responsible for coordinating the implementation of the change,
- the supervisor responsible for the implementation,
- any testing required by type and scope,
- any instructions and/or procedures requiring revision, and
- who will perform the actual physical implementation.

Upon completion of the change(s), a detailed RFC Final Summary Report must be prepared and an RFC Release and Review Form initiated. The RFC Final Summary Report is used to ensure that all items identified on the RFC Installation Authorization have been completed and the drawings used during the operation of the plant have been revised to reflect the change. The RFC Release and Review Form is used to process the RFC to closeout and to secure signatures necessary to declare the change operable and available for unrestricted use.

This description highlights the change control process. The attached flowchart provides further details.

ATTACHMENT 4 (Cont'd.)



ATTACHMENT 5 to AEP:NRC:0838A

VENDOR DOCUMENT CONTROL SYSTEM (VDCS)

We are in the process of setting up a computerized Vendor Document Control System. All technical information received for the D.C. Cook Nuclear Plant will be processed through this system regardless of whether the vendor sent the information to the plant or to the Corporate office. It is currently envisioned that the VDCS will operate in the following manner:

A central group will provide a log-in procedure which will allow for systematic storage and retrieval of documents by selected fields. Any document that does not pertain to Cook will terminate here. Applicable documents will be sent to the Cognizant Engineer for disposition. Where more than one discipline is involved, a response coordinator will be named with the responsibility of ensuring that proper action is taken and the computer records updated.

Transmittals between the Central group, the Engineering Divisions of the AEP Service Corporation, and the Plant will be tracked by traveler forms. Any plant or design change initiated by a vendor document will be referenced by its Plant Modification (PM) or Request For Change (RFC) number. Existing Plant and Corporate Procedures take over for the control of changes.

Periodically, all vendors on the Vendor Document Control System will be formally contacted and requested to furnish a list of all technical documents sent to AEP or the Cook Plant. This list will be compared to the computer list. Any documents missing from the VDCS will be obtained from the vendor and processed in the normal manner.

The Corporate Procedure is being drafted and the central group and computer support personnel have been identified. The implementation details have yet to be worked out. INPO is working on the problem from the generic standpoint of what documents are to be controlled and how to garner vendor support.

Since this response is in part dependent on input from an outside organization, i.e., INPO, it is not yet possible to develop an implementation schedule. We anticipate that we will be able to provide that implementation schedule on March 30, 1984.

ATTACHMENT 6 TO AEP:NRC:0838A
REACTOR TRIP SYSTEM RELIABILITY
WESTINGHOUSE BULLETINS BEING IMPLEMENTED:
RELEVANT PLANT PROCEDURES BEING REVIEWED FOR UPDATE

Number	Bulletin or Data Letter	Description
79-04		Solid State Protection System P-4 Permissive Test
80-03	D	Nuclear Instrumentation Connectors - Amphenol
80-5	B	Delta-Flux Instrumentation Calibration
80-07	D	Foxboro Transmitter Junction Box
80-8	B	Source Range Detector
73-20	B	Multi-Turn Potentiometers
74-3	D	Foxboro Transmitters-Calibration
77-3	D	Solid State Protection System - Extender Boards
80-8	D	Barton Transmitter Potentiometer Oxide Deposit
80-6	B	Being addressed by Request For Change (RFC) 12-2512