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

SUBJECT: Forwards response to NRC 840913 request for addl info re electrical equipment environ qualification program, consisting of justifications for continued operation.

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INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

October 18, 1984
AEP:NRC:0775N

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
ELECTRICAL EQUIPMENT ENVIRONMENTAL QUALIFICATION (10 CFR 50.49);
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

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

Dear Mr. Denton:

By letter dated September 13, 1984 [Mr. S. A. Varga, NRC, to Mr. John E. Dolan, Indiana & Michigan Electric Company (IMECo)], your staff requested additional information on the Donald C. Cook Nuclear Plant electrical equipment environmental qualification program. The requested information is contained in this letter and its attachments. It is understood that this information will form the basis for your staff's final review of the Cook Nuclear Plant 10 CFR 50.49 environmental qualification program, and allow for the issuance of a Safety Evaluation Report on this topic.

In particular, your staff requested that we submit all of the applicable Justifications for Continued Operation (JCOs) that are currently being relied upon. These JCOs are provided in Attachment 2 to this letter, and the bases for determining what constitutes an acceptable JCO are described in Attachment 1 to this letter. The JCOs provided herein supersede those previously provided in IMECo letter No. AEP:NRC:0775E, dated June 12, 1984.

Information has also been requested on our methodology for identifying those equipment items within the scope of 10 CFR 50.49(b)(2), on the design basis events which have been considered during our review, and on our treatment of Regulatory Guide 1.97 equipment items within the 10 CFR 50.49 qualification program. Our responses are provided in Attachment 1 to this letter.

It is also noted that the Westinghouse Owners Group (WOG) has recently undertaken a preliminary study to address the issue of potential superheated steam releases outside containment, resulting from a main steam line break with steam generator U-tube uncovery, and is currently evaluating the feasibility of conducting additonal studies. The preliminary assessment of this issue's impact on the Cook Nuclear Plant qualification program has previously been provided in IMECo letter No. AEP:NRC:0775M, dated August 3, 1984. Should the ongoing WOG study indicate the need for changes to our environmental

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qualification program, we will notify the NRC under the provisions of 10 CFR 50.49(h). During the interim period prior to final resolution by the WOG, however, we understand that the issue of superheated steam releases will not be explicitly referenced in the final Safety Evaluation Report. We therefore expect to resolve this issue on the schedule set by the WOG, rather than on the schedule for environmental qualification presented in 10 CFR 50.49(g).

Additionally, it is noted that one of the deficiencies identified in the Franklin Research Center Technical Evaluation Report pertained to the environmental qualification of Mobil greases and oils. We were informed by NRC staff during a meeting on September 13, 1983, that the data sheets provided by the vendor as proof of qualification would not be accepted without supporting documentation showing where the data sheet values came from. Our understanding, based on verbal communication with NRC staff, is that if supporting proprietary information is difficult to obtain, documentation indicating that we have audited the vendor's records for acceptability would constitute proof of qualification.

Based on information we had available in May 1984, we believed that an audit of Mobil would be confirmatory in nature and would not require a request for extension of qualification deadline and/or a JCO. The audit of Mobil Oil Corporation was conducted on June 7 and 8, 1984, to verify the adequacy of environmental testing of various lubricants. The documentation presented for inspection during the audit was deemed inadequate to substantiate Mobil's greases and oils qualification without additional documentation. Mobil Oil Corporation responded to our request for additional information by letter dated September 7, 1984, and indicated that, while testing data for various Mobil greases and oils presently in use is not available, it is their judgement that the current formulations would perform as required. We respect the considered judgement of Mobil Oil Corporation and agree that the lubricant can be considered acceptable. We understand, however, that Mobil does not intend to provide for any future testing of radiation resistance of their various grease formulations due to the limited commercial market. Therefore, a program for replacment of Mobil greases and oils with those where qualification data is more readily available is being developed for implementation.

Furthermore, during the preparation of the JCOs provided in Attachment 2 to this letter, it was determined that additional qualification deadline extension requests should be filed in accordance with 10 CFR 50.49(g). These extension requests are as follows:

- It has been recently discovered that the cables for Resistance Temperature Detectors (RTDs) Nos. NTR-110, -120, -130, -140, -210, -220, -230, and -240 have been inadvertently routed below the computed maximum containment flood level elevation. These cables will be rerouted above this elevation, but the work is required to be performed while the plant is shut down. The next outage of sufficient duration to perform this work is scheduled to take place during the

1985 refueling outages which are currently anticipated to begin in March 1985 for Unit 1 and November 1985 for Unit 2. Since this recent finding can impact our Inadequate Core Cooling Instrumentation, we are currently reviewing our previous submittals in this area to determine if this discovery requires an amendment to our earlier statements.

- It has been recently discovered that the triax instrument cables for the post-accident high range area monitors Nos. VRA-1310, -1410, -2310, and -2410 are not protected by flood-up tubing. Conax Corporation has been contacted for the design of a containment triax penetration flood-up tubing installation, and we have been informed that the special feedthrough penetrations will be delivered by February 1985. We therefore expect to perform the installation during the 1985 refueling outages which are currently anticipated to begin in March 1985 for Unit 1 and November 1985 for Unit 2. We are currently reviewing our previous submittals on NUREG-0737, Item II.F.1, to determine if this discovery requires an amendment to our earlier statements.

- We have previously committed to the installation of Conax connectors on the Reactor Coolant System vent system solenoid valves, in order to ensure that the installed plant configuration is as similar as possible to the tested configuration. Our current schedule for procurement and design will not guarantee an installation during the Unit No. 2 ice basket surveillance outage, scheduled to take place later this year. Therefore, we expect to perform the installation during the 1985 refueling outages which are currently anticipated to begin in March 1985 for Unit 1 and November 1985 for Unit 2.

- In order to partially resolve your staff's concerns with regard to the submergence qualifications for ITT Barton transmitters, we are planning to relocate the wide range Reactor Coolant System pressure transmitters to a location outside containment. This relocation is currently scheduled to take place during the 1985 refueling outages which are currently anticipated to begin in March 1985 for Unit 1 and November 1985 for Unit 2.

- Calculations made by Impell Corporation in their Aging Evaluation Program indicated an estimated thermal aging life between 10.9 and 40 years for the Westinghouse electric motors used as centrifugal charging, safety injection, and residual heat removal pump motors at the Cook Nuclear Plant. This was based on motor service temperatures higher than normal, and on a Westinghouse aging report written for motors of a later vintage than the Cook Nuclear Plant motors. To establish the required aging qualification, a purchase order has been issued to Westinghouse to perform the following services: (1) a material search of the materials contained in the Cook Nuclear Plant pump motors; (2) a definition as to the applicability of the

Westinghouse report to the Cook Nuclear Plant pump motors; and (3) a program to establish the pump motors' qualified life based on the results of items (1) and (2) above. Westinghouse expects to have items (1) and (2) complete by February 1985 and item (3) by July 1985.

This letter and its attachments have not yet been reviewed and approved by the Plant Nuclear Safety Review Committee (PNSRC) and the Nuclear Safety Design Review Committee (NSDRC). The review by both committees is scheduled to take place in the near future. If those reviews result in any changes to this letter, we will notify you.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich 10/18/84
Vice President

MPA/dam

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

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ATTACHMENT 1 TO AEP:NRC:0775N
RESPONSE TO SEPTEMBER 13, 1984, NRC REQUEST FOR INFORMATION
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

Question 1:

Submit all applicable JCO's that are currently being relied upon and confirm the following for each JCO associated with equipment that is assumed to fail:

> No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event will occur.

Response to Question 1:

Attachment 2 to this letter contains all of the Justifications for Continued Operation (JCOs) which are currently being relied upon. These JCOs supersede those previously provided in Attachment 1 to IMECo letter No. AEP:NRC:0775E, dated June 12, 1984.

All of the JCOs contained in Attachment 2 to this letter are based on the provisions of 10 CFR 50.49(i); that is, a JCO is considered acceptable if one or more of the following conditions have been met. It is also noted that each equipment item for which a JCO has been written need not necessarily be assumed to fail under adverse environmental conditions.

- The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified;

- The validity of partial type test data in support of the original qualification has been considered;

- There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified;

- The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator; or

- Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

Based on our engineering evaluation, we believe that postulated failure of equipment under the accident environment resulting from a design basis event will not lead to significant degradation of any safety function.

Question 2:

The licensee should confirm that in performing its review of the methodology to identify equipment within the scope of 10 CFR 50.49(b)(2) that the following steps have been addressed:

a. A list was generated of safety-related electric equipment as defined in paragraph (b)(1) of 10 CFR 50.49 required to remain functional during or following design-basis Loss of Coolant Accidents. The LOCA/HELB accidents are the only design-basis accidents which result in significantly adverse environments to electrical equipment which is required for safe shutdown or accident mitigation. The list was based on reviews of the Final Safety Analysis Report (FSAR), Technical Specifications, Emergency Operating Procedures, Piping and Instrumentation Diagrams (P&IDs), and electrical distribution diagrams;

b. The elementary wiring diagrams of the safety-related electrical equipment identified in Step a were reviewed to identified (sic) any auxiliary devices electrically connected directly into the control or power circuitry of the safety-related equipment (e.g., automatic trips) whose failure due to postulated environmental conditions could prevent required operation of the safety-related equipment and;

c. The operation of the safety-related systems and equipment were reviewed to identify any directly mechanically connected auxiliary systems with electrical components which are necessary for the required operation of the safety-related equipment (e.g., cooling water or lubricating systems). This involved the review of P&IDs, component technical manuals, and/or systems descriptions in the FSAR.

d. Nonsafety-related electrical circuits indirectly associated with the electrical equipment identified in Step a by common power supply or physical proximity were considered by a review of the electrical design including the use of applicable industry standards (e.g., IEEE, NEMA, ANSI, UL, and NEC) and the use of properly coordinated protective relays, circuit breakers, and fuses for electrical fault protection.

Response to Question 2.a:

The methodology used to identify equipment within the scope of 10 CFR 50.49(b)(1) for the Cook Nuclear Plant was essentially the same as that used to develop the IE Bulletin No. 79-01B equipment list. This is because the 10 CFR 50.49(b)(1) list was derived by reviewing the applicability of the IE Bulletin No. 79-01B list to the requirements of the environmental qualification rule.

As described in IMECo letter No. AEP:NRC:0775E referenced above, the methodology used to develop the 10 CFR 50.49(b)(1) equipment list via the IE Bulletin No. 79-10B equipment list was as follows:

(1) The IE Bulletin No. 79-01B list which used the process identified in this Step (1) was reviewed. The IE Bulletin No. 79-01B process developed a list of potentially affected equipment compiled primarily using the FSAR Chapters 5, 6, 7, and 14, Appendices O and Q to the FSAR, Cook Plant Emergency Operating Procedures, and previous licensing submittals to the NRC. Additional guidance for the IE Bulletin No. 79-01B listing was developed to define potentially affected areas of the Cook Plant and the postulated environment at various locations. The flow diagrams and the Technical Specifications were also reviewed when necessary to assure accuracy. The methodology used to develop the IE Bulletin No. 79-01B equipment list is also described in Attachment 6 to IMECo letter No. AEP:NRC:0578B, dated June 11, 1982.

(2) Using this IE Bulletin No. 79-01B listing, a review was performed using the following screening criteria in order to develop the 10 CFR 50.49(b)(1) equipment list:

    (a) Was the device located in an area subject to a LOCA, HELB, or MSLB accident environment?

    (b) Was the device required to function for a particular accident or during the accident duration?

    Equipment which did not meet these criteria was deleted from the equipment.list developed in Step (1).

(3) The list of equipment developed from Step (2) was then reviewed against NUREG-0578 and NUREG-0737. If this review indicated that installed electrical equipment should be added to the list developed in Step (2), the list was revised to reflect the change.

(4) The list of equipment developed from Step (3) was then reviewed to identify the associated electrical equipment that is located in a harsh environment (e.g., cables, terminations, etc.) servicing the primary equipment. The primary source of information was the cable

schematic diagrams. If more detail was required, the appropriate wiring diagrams and/or detail design drawings were consulted, as described in Attachment 3 to IMECo letter No. AEP:NRC:0775E referenced above.

It is believed that the above stated methodology, although different than the methodology specified in Question 2.a, adequately identified equipment within the scope of 10 CFR 50.49(b)(1).

Response to Question 2.b:

The elementary wiring diagrams of the safety related electrical equipment within the scope of 10 CFR 50.49 were reviewed to identify any auxiliary devices electrically connected directly into the control or power circuitry of the safety related equipment (e.g., automatic trips), whose failure due to postulated environmental conditions could prevent required operation of the safety related equipment. No 10 CFR 50.49(b)(2) equipment has been found in the 10 CFR 50.49(b)(1) equipment control circuits.

Response to Question 2.c:

The operation of safety related systems and equipment was reviewed to identify any mechanically connected auxiliary systems with electrical components that are necessary for the required operation of the safety related equipment. The primary documents used for this review were the flow diagrams. The equipment list submitted to identify equipment within the scope of IE Bulletin No. 79-01B reflects this review. The 10 CFR 50.49(b)(1) equipment list was developed as described in the "Response to Question 2.a" section above. If any auxiliary equipment contained in the IE Bulletin No. 79-01B equipment list met the subsequent requirements test delineated in the "Response to Question 2.a" section above, then it was also included in the 10 CFR 50.49 equipment list.

Response to Question 2.d:

Nonsafety related electrical circuits indirectly associated with the electrical equipment in the scope of 10 CFR 50.49 by common power supply or physical proximity were considered by a review of the electrical design, including the use of properly coordinated protective relays, circuit breakers, and fuses for electrical fault protection. No 10 CFR 50.49(b)(2) equipment has been found whose failure would cause a failure of the safety related equipment within the scope of 10 CFR 50.49. The circuit breaker and fuse coordination study, and the thermal coordination for the adequate protection of load conductors on 600-volt motor control centers study, were undertaken as a consequence of the 10 CFR 50 Appendix R fire protection requirements.

Question 3:

Provide confirmation that all design basis events which could potentially result in a harsh environment, including flooding outside containment, were addressed in identifying safety-related electrical equipment within the scope of 10 CFR 50.49(b)(1).

Response to Question 3:

To the best of our knowledge, all design basis events which could result in a harsh environment, including flooding outside containment, were addressed in the identification of electrical equipment within the scope of 10 CFR 50.49. As previously stated in IMECo letter No. AEP:NRC:0775E referenced above:

> ". . .The flooding and environmental consequences of the postulated design basis events documented in Chapter 14 of the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR), including the Loss-of-Coolant Accident (LOCA) and the Main Steam Line Break (MSLB) inside containment, were considered in specifying the qualification requirements for 10 CFR 50.49(b)(1) equipment. The environmental consequences of High Energy Line Breaks (HELBs) outside containment were also taken into account. With regard to flooding outside containment, we note that this topic was discussed in Attachment 10 to our letter No. AEP:NRC:0578B. . . .", referenced above.

Our treatment of the issue regarding potential superheated steam releases outside containment (resulting from a steam line break with U-tube uncovery) is described in the cover letter to this submittal.

Question 4:

Confirm that the electrical equipment within the scope of 10 CFR 50.49(b)(3) is all R.G. 1.97 Category 1 and 2 equipment or that justification has been provided for any such equipment not included in the environmental qualification program.

Response to Question 4:

As indicated in Attachment 3 to IMECo letter No. AEP:NRC:0775C, dated May 20, 1983, equipment which may fall into Regulatory Guide 1.97 Categories 1 and 2 will be reviewed under the requirements of Generic Letter No. 82-33, which provides for incorporation of this review into the integrated program established by NUREG-0737 Supplement 1. This position was also expressed in IMECo letter No. AEP:NRC:0775F, dated September 26, 1983.

Additionally, it has been noted that if our review of equipment for Regulatory Guide 1.97 requirements results in the discovery of any qualification concerns, then the provisions of 10 CFR 50.49(h) will be applied. This position has been accepted by letter dated October 24, 1983 [Mr. H. R. Denton (NRC) to Mr. John E. Dolan (IMECo)].

ATTACHMENT 2 TO AEP:NRC:0775N
JUSTIFICATIONS FOR CONTINUED OPERATION
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

This attachment contains the following JCOs for the Cook Nuclear Plant:

(1) Eberline DA1-HT-6CC Radiation Monitoring System Detector

(2) ITT Barton 763 Pressure Transmitter

(3) ITT Barton 764 Differential Pressure Transmitter

(4) ITT Barton 764 Differential Pressure Transmitters, Lot Nos. 1 and 2

(5) Raychem and Brand Rex Item Nos. 3074 and 3112 Triax Instrument Cable

(6) Reactor Coolant System Vents Solenoid-Operated Valves' Cable Terminations

(7) Samuel Moore and Boston Insulated Wire Item No. 3075 Instrument Cable

(8) Samuel Moore, Boston Insulated Wire, and Cerro Wire & Cable Item Nos. 3075 and 3077 Cables

(9) Victoreen 877-1 Radiation Monitoring System Detector

(10) Westinghouse Pump Motor Model Nos. 5808Z, 5009H, and 5009-P24

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 & 2

Equipment Manufacturer: Eberline

Equipment Model/Item No(s).: DA1-HT-6CC (see note next page)

Equipment Description: Radiation Monitoring System Detector

System Component Evaluation Worksheet No(s).: N/A

Plant Identification No(s).: VRS-1101,1201,2101,2201

Outstanding Equipment Qualification Deficiencies: SCEW sheets not written and test report not yet completely reviewed.

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

__X__(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

SCEW sheets have not yet been written and the test report has not been completely reviewed and submitted to the CEEOF. This will be completed by March 31, 1985. Preliminary reviews of the test report indicated that the subject equipment is qualified for the intended service.

It should be noted that the model number provided on the previous page is a correction to that submitted as part of the 10 CFR 50.49(b)(1) list. Originally listed as model number DA1-6CC.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 & 2

Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 763

Equipment Description: Pressure Transmitter

System Component Evaluation Worksheet No(s).: I22,I23 (Unit No. 1) and I23,I24 (Unit No. 2)

Plant Identification No(s).: NPS-121,122

Outstanding Equipment Qualification Deficiencies: Submergence

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

__x__(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

Based on telephone conversations with Westinghouse, American Electric Service Corporation personnel have learned that submergence testing of Model 763 transmitters has taken place. The applicability of this testing to the transmitters installed in the Donald C. Cook Nuclear Plant is still under evaluation.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 & 2

Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 764

Equipment Description: Differential Pressure Transmitter

System Component Evaluation Worksheet No(s).: N/A

Plant Identification No(s).: NLA-310, NLI-311, 320,321, 110,111,120,121, 130,131

Outstanding Equipment Qualification Deficiencies: SCEW sheets not written and test report not yet completely reviewed.

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

__x__(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

SCEW sheets have not yet been written and the test report has not been completely reviewed and submitted to the CEEOF. This will be completed by March 31, 1985. Preliminary reviews of the test report indicated that the subject equipment is qualified for the intended service.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 and 2

Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 764 Lot 1; 764 Lot 2

Equipment Description: Differential Pressure Transmitter

System Component Evaluation Worksheet No(s).: I1, I12, I18 (Unit No. 1); I1, I2, I13, I19 (Unit No. 2)

Plant Identification No(s).: (See next page for listing.)

Outstanding Equipment Qualification Deficiencies: Aging

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

__x__(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

A draft report on material aging has been prepared by Impell Corporation for these transmitters. The validity of the life values provided by Impell are, however, considered questionable by American Electric Power Service Corporation personnel. These values are therefore undergoing review in order to achieve resolution. In any event, the only materials of concern identified by the Impell report are the transmitter gasket materials.

The plant identification numbers covered by this JCO are the following:

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

MFC-110, -111, -120, -121, -130, -131, -140, and -141;

NLP-151, -152, and -153.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 and 2

Equipment Manufacturer: Raychem, Brand Rex

Equipment Model/Item No(s).: 3112, 3074

Equipment Description: Triax Instrument Cable for the Post Accident High Range Area Monitors VRA-1310, -1410, -2310, -2410

System Component Evaluation Worksheet No(s).: CI-13 - 15, for Unit 1; CI-21 - 22 for Unit 2

Plant Identification No(s).: N/A

Outstanding Equipment Qualification Deficiencies: Submergence

Justification For Continued Operation (check one or more of the following, and explain on the next page):

X (a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

____(b) The validity of partial type test data in support of the original qualification has been considered.

____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

The Radiation Monitoring System is a Post-TMI lessons learned installation. Same information can be obtained from the Post Accident Sampling System and from the present Lower Containment Radiation Monitor, after appropriate post-accident isolation actuation by the operator.

Since the Post Accident High Range Area Monitor has a long-term operation after an accident, its electrical cables have to be protected against submergence with flood-up tubing. Conax Corporation has been contacted for the design of a containment triax penetration flood-up tubing installation and a plant design change is planned.

Unit 1 installation should be performed during the 1985 refueling outage, currently scheduled to begin in March 1985; Unit 2 installation should be performed during the 1985 refueling outage, scheduled to begin in November 1985.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: Units 1 and 2

Equipment Manufacturer: N/A

Equipment Model/Item No(s).: N/A

Equipment Description: Cable termination at Reactor Coolant System vents solenoid operated valve

System Component Evaluation Worksheet No(s).: TC-16

Plant Identification No(s).: N/A

Outstanding Equipment Qualification Deficiencies: Steam and chemical spray

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

_____(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

__X__(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

The RCS vents are a Post-TMI lessons learned requirement. Equipment installation without proof of qualification was NRC mandated. When the environmental qualification test configuration details became available, Spring of 1984, we realized our installation did not conform with the qualified installation.

In their qualification test, Westinghouse used a Conax connector to preclude the possibility of steam and chemical spray from entering the solenoid enclosure and reacting with the electrical terminations. Since we had not been made aware of the details of the test configuration, the solenoid valves were installed with a metal conduit with no protection against steam environment.

To change the electrical connections to agree with those used by Westinghouse in their qualification test, a plant design change is in process.

The electrical connections should be modified before 3/31/85 on Unit 1 and during the next outage of sufficient duration on Unit 2.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 and 2

Equipment Manufacturer: Samuel Moore; Boston Insulated Wire

Equipment Model/Item No(s).: 3075

Equipment Description: Instrument Cable

System Component Evaluation Worksheet No(s).: Samuel Moore : CI-3, CI-5, Boston Insulated Wire : CI-5, CI-7

Plant Identification No(s).: Various

Outstanding Equipment Qualification Deficiencies: Submergence

Justification For Continued Operation (check one or more of the following, and explain on the next page):

X (a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

____(b) The validity of partial type test data in support of the original qualification has been considered.

____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

These cables serve the ITT Barton pressure transmitters with the following plant identification numbers:

Pressure transmitters: NPS-121, -122

These pressure transmitters will be relocated outside the reactor containment, thus eliminating the need for the instrument cable to be qualified for submergence. Relocation will take place during Unit 1 refueling outage scheduled for March, 1985 and Unit 2 refueling outage scheduled for November of 1985.

Should the cable serving the pressure transmitter instruments NPS-121 and -122 fail, the long term post accident function can be provided by several alternate devices either not subject to flooding or not located inside containment. For MSLB and small LOCA incidents, these devices are backed up by the pressurizer pressure NPP transmitters. Since the concern for MSLB is pressurization and for small LOCA is pressure hang-up, the 1700 psig low range portion of the NPPs is adequate. For intermediate or large LOCA incidents, RCS pressure may be obtained by determining which of the ECCS pumps are actually delivering flow and at what pump discharge pressure. The same method may be used for MSLB cooldown concerns.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

Therefore, the capability of the plant to safely shutdown is not impeded.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 and 2

Equipment Manufacturer: S. Moore, Boston Insulated Wire, Cerro Wire & Cable

Equipment Model/Item No(s).: 3075, 3077

Equipment Description: Cables for temperature detectors (NTR's) in cold leg and hot leg steam generators

System Component Evaluation Worksheet No(s).: CI-3 and CI-5 for Unit 1 CI-8, CI-9, CI-11 for Unit 2

Plant Identification No(s).: NA

Outstanding Equipment Qualification Deficiencies: Submergence

Justification For Continued Operation (check one or more of the following, and explain on the next page):

__x__(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

_____(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

The NTR's -110, -120, -130, -140, -210, -220, -230, and -240 are temperature detectors located in the cold and hot leg piping of each loop, that measure hot and cold leg coolant temperatures. The same information can be obtained from the Main Steam Pressure Transmitters in conjunction with saturated steam tables.

Routing of the NTR cables inadvertently brought these cables below the flood level elevation within containment. They will be rerouted above the maximum flood level elevation. A design change has been issued to cover this work which should be completed during the 1985 refueling outages for each Unit of the Donald C. Cook Nuclear Plant (i.e., those outages anticipated to begin in March 1985 for Unit 1 and in November 1985 for Unit 2).

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 & 2

Equipment Manufacturer: Victoreen

Equipment Model/Item No(s).: 877-1

Equipment Description: Radiation Monitoring System Detector

System Component Evaluation Worksheet No(s).: N/A

Plant Identification No(s).: VRA-1310, 1410, 2310, 2410

Outstanding Equipment Qualification Deficiencies: SCEW sheets not written and test report not yet completely reviewed.

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

__X__(b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

SCEW sheets have not yet been written and the test report has not been completely reviewed and submitted to the CEEOF. This will be completed by March 31, 1985. Preliminary reviews of the test report indicated that the subject equipment is qualified for the intended service.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Donald C. Cook Nuclear Plant No(s).: 1 and 2

Equipment Manufacturer: Westinghouse

Equipment Model/Item No(s).: 5808Z, 5009H, 5009-P24

Equipment Description: Centrifugal charging, safety injection, and residual heat removal pump motors

System Component Evaluation Worksheet No(s).: M-1

Plant Identification No(s).: PP-050, -026, -035

Outstanding Equipment Qualification Deficiencies: Aging

Justification For Continued Operation (check one or more of the following, and explain on the next page):

_____(a) The safety function may be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

X (b) The validity of partial type test data in support of the original qualification has been considered.

_____(c) There is limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

_____(d) The safety function will be completed prior to exposure to the accident environment resulting from the design basis event, and subsequent failure of the equipment will not degrade any safety function or mislead the operator.

_____(e) Failure of the equipment under the accident environment resulting from the design basis event will not lead to significant degradation of any safety function or misleading information to the operator.

JUSTIFICATION FOR CONTINUED OPERATION (continued):

Explanation of Justification For Continued Operation Noted on Previous Page:

Calculations made by Impell Corporation in their Aging Evaluation Program indicated an estimated thermal aging life of between 10.9 and 40 years for the Westinghouse electric motors used as centrifugal charging, safety injection, and residual heat removal pump motors installed at the D.C. Cook Nuclear Plant Units 1 and 2. This was based on motor service temperatures higher than normal, and on a Westinghouse aging report written for motors of a later vintage than the D.C. Cook Plant motors. To establish the required aging qualification, a Purchase Order has been issued for Westinghouse to perform the following services: 1) A material search of the materials contained in the D.C. Cook pump motors; 2) A definition as to the applicability of the Westinghouse report to the D.C. Cook Plant motors; 3) A program to establish the pump motors qualified life based on the results of items (1) and (2) above. Westinghouse expects to have Items (1) and (2) complete by February, 1985 and Item (3) by July, 1985.