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ACCESSION NBR:8406180370 DOC.DATE: 84/06/12 NOTARIZED: NO DOCKET #
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50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards justification for continued operation in support of environ qualification of electric equipment program,per request.

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[illegible] for continuous operation [illegible]

[illegible] qualification of electric [illegible]

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[illegible]

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

June 12, 1984
AEP:NRC:0775E

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT (10 CFR 50.49);
JUSTIFICATION FOR CONTINUED OPERATION

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

Dear Mr. Denton:

This letter provides additional information in support of the Donald C. Cook Nuclear Plant 10 CFR 50.49 environmental qualification program, as requested during telephone conversations with your staff. More specifically, Attachment 1 to this letter provides our Justifications for Continued Operation (JCOs) for certain electric equipment items. In most cases, we have adapted these JCOs from the provisions of 10 CFR 50.49(i).

Attachment 2 to this letter contains a copy of a Westinghouse letter regarding Resistance Temperature Detector (RTD) chemical spray parameters used in WCAP-9157. This Westinghouse letter has been provided in support of JCO No. 6 (included in Attachment 1 to this letter).

Attachment 3 to this letter provides a description of the methodology used to identify those electric equipment items within the scope of 10 CFR 50.49(b)(2), i.e., those items which are non-safety related but important to safety, as defined by the final rule on environmental qualification. It should be noted that this methodology was also used to identify those electric equipment items within the scope of 10 CFR 50.49(b)(1) and 10 CFR 50.49(b)(3), i.e., safety related equipment and certain post-accident monitoring equipment, respectively.

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With regard to your staff's verbal request that we describe our treatment of Regulatory Guide 1.97 equipment items, we note that our plans for the implementation of 10 CFR 50.49 requirements have been delineated in Attachment 3 to our letter No. AEP:NRC:0775C, dated May 20, 1983, and in our letter No. AEP:NRC:0775F, dated September 26, 1983.

During our conversations with your staff, we were requested to confirm that all design basis events at the Donald C. Cook Nuclear Plant which could result in a harsh environment, including flooding outside containment, were addressed in the identification of safety related electrical equipment requiring environmental qualification to the provisions of 10 CFR 50.49. 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, dated June 11, 1982. We also note that since safe shutdown for the Donald C. Cook Nuclear Plant is hot shutdown under the design basis, we do not interpret 10 CFR 50.49 as necessarily requiring the qualification of equipment needed to achieve and maintain cold shutdown conditions.

Additionally, Section IV of Attachment 1 to our letter No. AEP:NRC:0775G, dated January 17, 1984, indicated that we would determine if Donald C. Cook Nuclear Plant valves ICM-111, ICM-129, and IMO-128 would be submerged by a feedwater line break inside containment and, if so, if they could properly function after an accident. We have completed our review of this issue and have determined that submergence of these valves will not prevent us from achieving and maintaining hot shutdown. In particular, we note that Section 4.3.4 of the Franklin Research Center Technical Evaluation Report for the Donald C. Cook Nuclear Plant states the following (where the names of systems appearing in parentheses have been added for clarity):

> ". . .IMO-128 and ICM-129 are the two in-series valves in the normal (Residual Heat Removal) RHR letdown line and ICM-111 is in the normal RHR cooldown return line. These valves are not part of the (Emergency Core Cooling System) ECCS and serve no safety function other than to maintain Reactor Coolant System (RCS) isolation when pressure is above the RHR design pressure. These valves are normally closed during operation. Although the motor operators for these valves would reasonably be expected to remain operational when subjected to an environment with a pH between 8.5 and 11.0, their failure to do so in such an environment does not adversely impact any safety analysis conclusions. . . ."

As noted in Section II of Attachment 1 to our letter No. AEP:NRC:0775G, dated January 17, 1984, we are conducting an ongoing audit of our environmental qualification documentation files. Some preliminary results of this audit are currently under study, and it is anticipated that we will request an amendment of our 10 CFR 50.49 equipment list. This request will be transmitted by the end of this month, along with a modified schedule for completion of our documentation files.

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
Vice President 6/12/84

MPA/dam

Attachments

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

[illegible]

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ATTACHMENT 1 TO AEP:NRC:0775E

JUSTIFICATIONS FOR CONTINUED OPERATION (JCOs)

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

This Attachment contains the JCOs for certain electric equipment items important to safety which are installed in the Donald C. Cook Nuclear Plant. The following points regarding these JCOs should be noted:

- For each JCO presented in this Attachment, an attempt has been made to correlate the JCO with one of the five (5) justifications presented in paragraph (i) of 10 CFR 50.49. We have, however, taken the liberty of justifying continued Plant operation on other than the five (5) justifications cited therein when we believed there was adequate cause (i.e., when further review indicated that previous conclusions regarding equipment inadequacy were unfounded).

- In IMECo's letter No. AEP:NRC:0775C, dated May 20, 1983, no electric equipment items were claimed to be completely qualified to the provisions of 10 CFR 50.49. This conservative position was taken primarily because the results of analyses regarding materials' susceptibility to thermal and radiation aging had not yet been completed in accordance with Section 7.0 of the Division of Operating Reactors (DOR) Guidelines. As noted in IMECo's letter No. AEP:NRC:0775G, dated January 17, 1984, American Electric Power Service Corporation (AEPSC) had contracted with a consultant to perform the required materials analyses and provide input for an eventual Surveillance/Maintenance/Replacement (SMR) program for electric equipment important to safety at the Donald C. Cook Nuclear Plant. The results of the materials analyses are currently being received by AEPSC where their impact on Plant operations are being assessed. At the present time, it is believed that only two (2) aging related concerns have been identified. More specifically, JCO No. 3 notes that a problem may exist with aging of ITT Barton Model No. 764 Lot 1 and Model No. 764 Lot 2 differential pressure transmitters. This issue is, however, still unresolved as we have questioned the bases for our consultant's conclusion. Until our consultant resolves this issue, we cannot be convinced that an aging problem exists, and thus the JCO can only serve to inform NRC staff of a potential problem. Additionally, JCO No. 7 notes that the gaskets on the Pressurizer Power Operated Relief Valves' (PORVs') NAMCO Model No. EA180 limit switch assemblies have achieved the end of their useful life. A design change schedule has been developed to resolve this issue at the earliest possible date consistent with our Plant outage schedules.

- Materials analyses have been completed for many electric equipment items within the scope of 10 CFR 50.49, although implementation of these analyses into a SMR program for the Donald C. Cook Nuclear Plant has not yet been completed. For those cases where indications exist that the installed life of an item may currently be in excess of the equipment's useful life, a JCO has been provided (see JCO Nos. 3 and 7, described above). If an equipment item has been determined to have a useful life in excess of its currently installed life, then it is believed that a JCO does not have to be provided for that equipment item, whether or not the item will eventually be added to the Donald C. Cook Nuclear Plant SMR program. This category also includes those equipment items, such as Foxboro Model No. E13DM-HSAH1 (MCA) differential pressure transmitters, for which our consultant states that regular maintenance and calibration will suffice for continued operation. Equipment items which fall into this category include, but are not necessarily limited to, the following:

  - ITT Barton Model No. 763 Pressure Transmitter; Plant Identification Nos. NPP-151, -152, -153, NPS-121, -122, -153; System Component Evaluation Worksheet (SCEW) Nos. I19, I20, I21, I22, I23, I24 (Unit No. 1), I20, I21, I22, I23, I24, I25 (Unit No. 2).

  - Rosemount Model No. 11834B or Sostman Model No. 176KF Resistance Temperature Detector (RTD); Plant Identification Nos. NTP-110, -111, -120, -121, -130, -131, -140, -141, -210, -211, -220, -221, -230, -231, -240, -241; SCEW Nos. I25, I26, I27 (Unit No. 1), I26, I27 (Unit No. 2).

  - Rosemount Model No. 11901B or Sostman Model No. 176KS RTD; Plant Identification Nos. NTR-110, -120, -130, -140, -210, -220, -230, -240; SCEW No. I28 (Unit Nos. 1 and 2). Note: These RTDs are to be replaced as part of the Donald C. Cook Nuclear Plant 10 CFR 50, Appendix R, Fire Protection Program, as previously stated in Attachment 4 to our letter No. AEP:NRC:0775C, dated May 20, 1983.

  - Foxboro Model No. E13DM-HSAH1 (MCA) Differential Pressure Transmitter; Plant Identification Nos. FFC-210, -211, -220, -221, -230, -231, -240, -241; SCEW Nos. I3 (Unit No. 1), I4 (Unit No. 2).

  - Foxboro Model No. E11GM-HSAE1 (MCA) Pressure Transmitter; Plant Identification Nos. MPP-210, -211, -220, -221, -230, -231, -240, -241; SCEW Nos. I14 (Unit No. 1), I15 (Unit No. 2).

  - Foxboro Model No. NE13DM-HIM1-D Differential Pressure Transmitter; Plant Identification Nos. FFI-210, -220, -230, -240; SCEW Nos. I4 (Unit No. 1), I5 (Unit No. 2).

- ASCO Model No. 206-381-2RVU Solenoid Valve; Plant Identification Nos. XSO-291, -292, -293, -294, -295, -296, -297, -298; SCEW No. S3 (Unit Nos. 1 and 2).

- ASCO solenoid valve Model No. NP-8316-54V (Plant Identification Nos. XSO-12, -21, -121, -122, -123, -124, -125, -126, -127, -320, -503, -505, and -507) has been reviewed for aging considerations. It is believed that the solenoid valve has a useful life of 40 years at 110°F, or 4.4 years at 140°F with the coil continuously energized. The valves have not yet operated for a total of 4.4 years and, therefore, have not yet reached the end of their useful life. These items will be rebuilt or replaced as required at a date consistent with our plant outage schedules.

- As previously stated in Section II of Attachment 1 to our letter No. AEP:NRC:0775G, dated January 17, 1984, we are pursuing the documentation of environmental qualification for Foxboro Model NE transmitters and Target Rock solenoid actuated globe valves. The required test reports have been received and a review is in progress to ensure full qualification. If our review indicates a qualification problem, we will submit JCOs and report the problem in accordance with 10 CFR 50.49(h).

- We are currently working with Mobil to obtain proper documentation of the environmental qualification for Mobilux EP2 lubricant, as previously stated in Section II of Attachment 1 to our letter No. AEP:NRC:0775G, dated January 17, 1984. We have written Mobil about confirmatory test data documenting that Mobilux EP2 meets the necessary environmental qualification requirements. We have been advised by phone that they will give us a written response. In the event a problem arises with respect to this confirmatory information, a JCO will be written.

The following pages contain the JCOs for certain electric equipment items which are installed in the Donald C. Cook Nuclear Plant. The standardized format for these JCOs includes the following information:

- JCO number.

- Donald C. Cook Nuclear Plant Unit number(s).

- The equipment item manufacturer(s), model or item number(s), and equipment item description.

- One or more SCEW numbers. These entries can be used to cross reference the JCOs to the qualification data in the Donald C. Cook Nuclear Plant Central Equipment Environmental Qualification File (CEEQF).

- Plant identification or tag number(s).

- Identification of the equipment environmental qualification deficiency of concern.

- One or more JCO, generally adapted from 10 CFR 50.49(i). Other reasons for JCO may be utilized by marking "Other".

- An explanation for the choice of JCO.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 1
Donald C. Cook Nuclear Plant Unit No(s).: 1 and 2
Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 764

Equipment Description: Differential Pressure Transmitter

System Component Evaluation Worksheet (SCEW) No(s).: I1 (Unit No. 1); I1, I2 (Unit No. 2)
Plant Identification No(s): BLP-112, -122, -132, -142

Outstanding Equipment Deficiencies: Submergence

Justification For Continued Operation (check one or more):

_____ (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 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.

__X__ (f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

There are three (3) transmitters that perform the level indication function on each of the four (4) Steam Generators (S/Gs). All transmitters have the same range. One (1) transmitter on each S/G is above the maximum floodup elevation of 614 ft. These transmitters are BLP-112, -122, -132, and -142, and are therefore not required to be qualified for submergence. The SCEWs which were previously

Justification For Continued Operation (JCO) No.: 1
(Continued)

transmitted to the NRC for these transmitters via letter No. AEP:NRC:0578B, dated June 11, 1982, showed these devices as submerged in order to reflect a conservative set of conditions for the group of like function equipment. It is now evident that a separate SCEW should have been developed for BLP-112, -122, -132, and -142 to reflect the fact that they are not submerged. The appropriate SCEWs will be revised and/or created to reflect the correct qualification requirements, but will not be resubmitted to the NRC.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 2
Donald C. Cook Nuclear Plant Unit No(s).: 1 and 2
Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 764

Equipment Description: Differential Pressure Transmitter

System Component Evaluation Worksheet (SCEW) No(s).: I1 (Unit No. 1); I1, I2 (Unit No. 2)
Plant Identification No(s): BLP-110, -111, -120, -121, -130, -131, -140, -141
Outstanding Equipment Deficiencies: Submergence

Justification For Continued Operation (check one or more):

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 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.

X (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.

___ (f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

As described in JCO No. 1, one (1) transmitter per S/G is located above the maximum flood elevation inside containment. The other eight (8) transmitters (two (2) per S/G) are located below the maximum flood elevation, at approximately the 601 ft elevation. These transmitters must perform an actuation function no later than 22.1 seconds into a Main Steam Line Break (MSLB). Because a flood

Justification For Continued Operation (JCO) No.: 2
(Continued)

elevation of 601 ft will not be reached within this time, submergence qualification of these transmitters is not required.

Donald C. Cook Nuclear Plant SCEW sheets indicate that these transmitters are to be used for long term post-accident monitoring. Again, as stated in JCO No. 1, these SCEW sheets were originally written in order to reflect a conservative set of conditions for the group of like function equipment. It is now apparent that BLP-110, -111, -120, -121, -130, -131, -140, and -141 are not required for long term post-accident monitoring. This is shown in our Plant Technical Specifications which require only one (1) transmitter per S/G to function for the purpose of long term post-accident monitoring. In addition, redundancy is accomplished by way of the auxiliary feedwater flow transmitters FFI-210, -220, -230, and -240 which are located outside containment and are fully qualified.

The applicable Plant procedures will be revised to indicate that, in the event of the transmitter at the highest elevation on any S/G becoming inoperable, entry into the Action statement of the Technical Specifications will occur. The appropriate SCEWs will be revised or developed to reflect the correct qualification, but will not be resubmitted to the NRC.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 3
Donald C. Cook Nuclear Plant Unit 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 (SCEW) No(s).: I1, I12, I18 (Unit No. 1); I1, I2, I13, I19 (Unit No. 2)
Plant Identification No(s).: (See second page of this JCO for list.)

Outstanding Equipment Deficiencies: Aging

Justification For Continued Operation (check one or more):

____(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 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.

____(f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

A draft report on material aging has been prepared by Impell for these transmitters. American Electric Power Service Corporation personnel have requested information from Impell pertaining to the data the report is based on, as supplemental aging data presented in Westinghouse Electric Corporation report No. NS-TMA-1950 (and subsequent reports) does not appear to have been taken into account. To this

Justification For Continued Operation (JCO) No.: 3
(Continued)

date, no reply has been received from Impell, and thus the validity of the life values provided by Impell are considered questionable. In any event, the only materials of concern identified by the Impell report are the transmitter gasket materials. Upon evaluation of any response from Impell, an expedited program would be effected, if necessary, to address any justified concerns.

Plant Identification Nos.: BLP-110, -111, -112,
-120, -121, -122,
-130, -131, -132,
-140, -141, -142

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

NLP-151, -152, -153

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 4
Donald C. Cook Nuclear Plant Unit No(s).: 1 and 2
Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 764

Equipment Description: Differential Pressure Transmitter

System Component Evaluation Worksheet (SCEW) No(s).: I12 (Unit No. 1); I13 (Unit No. 2)

Plant Identification No(s).: MFC-110, -111, -120, -121, -130, -131, -140, -141
Outstanding Equipment Deficiencies: Submergence

Justification For Continued Operation (check one or more):

_____(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 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.

__X__(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.

_____(f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

These transmitters must perform their safety function within five (5) seconds of a Main Steam Line Break accident. Since the containment will not flood up to Elevation 600 ft (i.e., the location of the transmitters) within the first five (5) seconds following a Main Steam Line Break, it is believed that this hardware will adequately perform its safety function prior to submergence.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 5
Donald C. Cook Nuclear Plant Unit No(s).: 1 and 2
Equipment Manufacturer: ITT Barton

Equipment Model/Item No(s).: 763

Equipment Description: Pressure Transmitter

System Component Evaluation Worksheet (SCEW) No(s).: I22, I23 (Unit No. 1); I23, I24 (Unit No. 2)
Plant Identification No(s).: NPS-121, -122

Outstanding Equipment Deficiencies: Submergence

Justification For Continued Operation (check one or more):

_____ (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 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.

_____ (f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

Based on telephone conversations with Westinghouse, American Electric Power 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)

Justification For Continued Operation (JCO) No.: 6
Donald C. Cook Nuclear Plant Unit No(s).: 1 and 2
Equipment Manufacturer: Rosemount or Sostman

Equipment Model/Item No(s).: 11834B, 11901B (Rosemount); 176KF, 176K (Sostman)
Equipment Description: Resistance Temperature Detector (RTD)

System Component Evaluation Worksheet (SCEW) No(s).: I25, I26, I27, I28 (Unit No. 1); I26, I27, I28 (Unit No. 2)
Plant Identification No(s).: (See second page of this JCO for list.)

Outstanding Equipment Deficiencies: Chemical Spray

Justification For Continued Operation (check one or more):

____ (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 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.

____ (f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

Westinghouse Electric Corporation report No. WCAP-9157, dated September 1977, did not explicitly identify the chemical spray conditions to which the RTDs were tested. As indicated in Attachment 2 to this letter, Westinghouse has reviewed their test records and verified the chemical spray parameters. This supplemental documentation will be included in our Central Equipment Environmental

Justification For Continued Operation (JCO) No.: 6
(Continued)

Qualification File as required by our Corporate procedures. No further action is anticipated.

Plant Identification Nos.: NTP-110, -111, -120, -121, -130, -131, -140, -141, -210, -211, -220, -221, -230, -231, -240, -241

NTR-110, -120, -130, -140, -210, -220, -230, -240

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 7
Donald C. Cook Nuclear Plant Unit No(s).: 1 and 2
Equipment Manufacturer: NAMCO

Equipment Model/Item No(s).: EA180

Equipment Description: Limit Switch

System Component Evaluation Worksheet (SCEW) No(s).: LS1

Plant Identification No(s).: Limit Switches For Pressurizer Power Operated Relief Valves (PORVs) NRV-151, -152, and -153
Outstanding Equipment Deficiencies: Aging

Justification For Continued Operation (check one or more):

_____ (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 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.

X (f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

The life of the NAMCO switch is effectively limited by the assembly cover gasket material to approximately 4 to 7 years, depending on the ambient temperature over the installed life of the switch. These gaskets will be replaced during the present refueling outage for Unit No. 2 and during the next refueling outage for Unit No. 1. It is

Justification For Continued Operation (JCO) No.: 7
(Continued)

also noted that these limit switches are not used in the control of the Pressurizer PORVs, but rather are used to indicate PORV position to the Control Room operator. In the event of an accident, the operator may close the Pressurizer PORVs' block valves to compensate for any uncertainties in the position of the PORVs.

JUSTIFICATION FOR CONTINUED OPERATION (10 CFR 50.49)

Justification For Continued Operation (JCO) No.: 8
Donald C. Cook Nuclear Plant Unit 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 (SCEW) No(s).: Samuel Moore: CI3, CI5; Boston Insulated Wire: CI5, CI7
Plant Identification No(s).: Various

Outstanding Equipment Deficiencies: Submergence

Justification For Continued Operation (check one or more):

_____ (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 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.

__X__ (f) Other (see explanation below).

Explanation Of Justification For Continued Operation Noted Above:

These cables serve the differential pressure transmitters and pressure transmitters discussed in JCO Nos. 1, 2, 4, and 5. These cables will either be relocated above the maximum floodup level within the containment, or qualified for 4 months of submergence. JCOs explained in JCOs 1, 2, and 4 also apply to the cable serving those instruments, i.e., the

Justification For Continued Operation (JCO) No.: 8
(Continued)

cable serving JCO No. 1 instruments will not be submerged, and the cable serving JCO No. 2 and 4 instruments will either not be submerged or will perform its safety function before being submerged.

Should the cable serving instruments listed in JCO No. 5 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 repressurization 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, reactor coolant system pressure may be obtained by determining which of the Emergency Core Cooling System pumps are actually delivering flow and at what pump discharge pressure. The same method may be used for MSLB cooldown concerns. Therefore, the capability of the plant to safely shut down is not impeded.

ATTACHMENT 2 TO AEP:NRC:0775E

RESISTANCE TEMPERATURE DETECTOR CHEMICAL SPRAY QUALIFICATION

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

file EQ

Westinghouse
Electric Corporation

Water Reactor
Divisions

Nuclear Services
Integration Division

Box 2728
Pittsburgh Pennsylvania 15230-2728

AEP-84-543

March 5, 1984

Mr. M. P. Alexich, Vice President
and Director Nuclear Engineering
American Electric Power Service Corporation
One Riverside Plaza
Columbus, OH 43216

Attn: W. G. Sotos

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D. C. COOK UNITS 1 AND 2
<u>RTD Chemical Spray Testing Parameters</u>

Dear Mr. Alexich:

In response to your letter AWS 101 regarding WCAP-9157 and the chemical spray parameters such as spray density and duration used in the testing, the following information is provided. Details of the chemical spray testing are provided in section 5-2 of the WCAP. That section includes the injection flow rate (6 GPH) and the weight percent of boric acid and sodium hydroxide which establish the PH of the chemical injection. A review of test records has confirmed the actual measured PH of the chemical injection to be 8.25 and the duration of the injection to be 24 hours.

If your have any further questions or require additional information, please contact me.

Very truly yours,

A. P. Suda for

W. J. Johnson, Manager
Customer Programs
Central Area

WJJ/dlc

cc: M. P. Alexich
W. G. Sotos
R. L. Shoberg

W. G. Smith
J. C. Jeffrey

0664f:12

ATTACHMENT 3 TO AEP:NRC:0775E

METHODOLOGY USED TO IDENTIFY EQUIPMENT

WITHIN THE SCOPE OF 10 CFR 50.49(b)(2)

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

Paragraph (g) of 10 CFR 50.49 requires, in part, that each holder of an operating license issued prior to February 22, 1983, identify the equipment important to safety within the scope of the final rule on environmental qualification of electric equipment. One category of equipment within the scope of the final rule is defined by 10 CFR 50.49(b)(2) as those items of nonsafety related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of the following:

- ensuring the integrity of the reactor coolant pressure boundary;
- ensuring the capability to shut down the reactor and maintain it in a safe shutdown condition; and
- ensuring the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR 100 guidelines.

The methodology that was used to identify electric equipment within the scope of 10 CFR 50.49(b)(2) for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2 follows:

(1) Starting with the equipment list developed for our IE Bulletin No. 79-01B submittals, a review was performed utilizing the screening criteria of step (2) below. The methodology used in developing the IE Bulletin No. 79-01B equipment list is described in Attachment 6 to our letter No. AEP:NRC:0578B, dated June 11, 1982. This review methodology was not repeated when performing the 10 CFR 50.49 review.

(2) The list developed in step (1) above was reviewed against two (2) criteria, namely:

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

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

  Equipment which did not meet the criteria above was deleted from the equipment list described in step (1) above.

(3) The list of equipment developed from step (2) was then reviewed against NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," dated July 1979, and NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980. If the review of these documents indicated that electric equipment presently installed in the Donald C. Cook Nuclear Plant should be added to the list of equipment developed from step (2) above, the list was revised to reflect this equipment. (As an example, it is noted that certain radiation monitors were added to the list during this phase of the review process.)

(4) The list of equipment developed in step (3) was provided to the American Electric Power Service Corporation Electrical Engineering Division for review. For every equipment item on the list developed in step (3) above, that auxiliary electric equipment installation (i.e., cable, cable terminations, limit switches, electrical penetrations, etc.) which is installed in a potentially harsh environment was identified and added to the appropriate SCEWs. Those items of electric equipment such as transformers, switching equipment, power sources, control switches, etc., which were determined to be installed in a mild environment were not added to the list as they are not within the scope of IE Bulletin No. 79-01B or 10 CFR 50.49.

(5) In determining the nature of the electrical installations under step (4) above, cable schematic diagrams indicating cable numbers, cable terminations, and electrical penetrations were used for each application. When more detailed information was required with regard to the electrical installation, the appropriate wiring diagrams and/or detail design drawings were consulted. Once the cable number was obtained, the cable and conduit schedule drawings were used to determine the cable item number (i.e., the cable size and type). The Purchase Order (PO) number and manufacturer were then determined for each cable item number from the "Cable Purchase Record Book" kept by the American Electric Power Service Corporation Electrical Engineering Division. In some cases, it was determined that more than one manufacturer was used to supply a given cable item number. In such cases, the qualification of each manufacturer's cable was reviewed to determine adequacy for the given electrical installation. If all possible cables were found to be qualified, the inquiry was considered complete. If, on the other hand, one or more possible cables were found to be deficient in their qualification, the source reel number for the cable was obtained from the cable pull cards maintained at the Donald C. Cook Nuclear Plant site. The reel number identifies the manufacturer and PO number for the installed cable, and thus the adequacy of the installed cable for environmental qualification purposes was obtained. In this fashion, for every cable number in the schematics diagrams, the exact cable size and manufacturer may be determined.

(6) Additionally, in compiling the list of equipment involved in the electrical installations under step (4), information contained in the cable schematics diagrams, wiring diagrams, qualification test reports, and Donald C. Cook Nuclear Plant installation practices allowed for the determination of the number and kind of cable terminations used on each installation. As an example, control cable installations with penetration extension wires placed inside floodup tubes are spliced at the penetrations inside the floodup tubes, at a floodup box near Elevation 614' (the maximum floodup level inside containment), and at the terminal boxes near the devices in question.

(7) It is noted that the 480 volt and 600 volt power circuits at the Donald C. Cook Nuclear Plant feeding equipment inside containment are equipped with redundant series circuit breakers. A short circuit caused by the harsh environment inside containment should always be interrupted by a circuit breaker feeding the damaged cable, even if one of the redundant circuit breakers fails to operate. The reason for the redundant breakers in the power circuits is to provide protection for the containment penetration seals within the requirements of the single failure criterion. Instrumentation and Control circuits are not equipped with redundant circuit breakers since the fault current availability is low and poses no threat to containment integrity. A short circuit on an instrument or control cable inside containment, caused by the harsh environment, will cause the protective device in a distribution cabinet outside containment (and in a mild environment) to trip and isolate the fault. If the protective device failed to trip, the other coordinated protective device (i.e., a circuit breaker) feeding the distribution cabinet will trip, isolating the fault but interrupting power to some safety related equipment at the same time. However, the redundant device on the redundant safety train will be available to perform the required safety function.

The methodology described above is believed to adequately address all electrical equipment within the scope of 10 CFR 50.49(b)(1), 10 CFR 50.49(b)(2), and 10 CFR 50.49(b)(3).

DISTRIBUTION
Docket File w/o encl.
ORB#1 Rdg w/o encl.
CParrish w/o encl.
DWigginton w/o encl.

JUN 6 1984

DOCKET NO(S).50-315/316

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43215

SUBJECT: INDIANA ANDMICHIGAN ELECTRIC COMPANY
Donald C. Cook Nuclear Plant

The following documents concerning our review of the subject facility are transmitted for your information.

- ☐ Notice of Receipt of Application, dated ________.
- ☐ Draft/Final Environmental Statment, dated ________.
- ☐ Notice of Availability of Draft/Final Environmental Statement, dated ________.
- ☐ Safety Evaluation Report, or Supplement No. ____, dated ________.
- ☐ Notice of Hearing on Application for Construction Permit, dated ________.
- ☐ Notice of Consideration of Issuance of Facility Operating License, dated ________.
- ☐ Monthly Notice; Applications and Amendments to Operating Licenses Involving no Significant Hazards Considerations, dated ________.
- ☐ Application and Safety Analysis Report, Volume ________.
- ☐ Amendment No. ____ to Application/SAR dated ________.
- ☐ Construction Permit No. CPPR- ____, Amendment No. ____ dated ________.
- ☐ Facility Operating License No. ____, Amendment No. ____, dated ________.
- ☐ Order Extending Construction Completion Date, dated ________.
- ☐ Other *(Specify)* Monthly Notice covering period throught May 23, 1984.
Expiration date for hearing requests and comments June 22, 1984.

Division of Licensing, ORB#1
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: w/enclosure

OFFICE➤	ORB#1:DL						
SURNAME➤	CParrish:ps						
DATE➤	6/6/84						

NRC FORM 318 (1/84) NRCM 0240

Indiana and Michigan Electric Company

Donald C. Cook Nuclear Plant, Units 1 and 2

cc: Mr. M. P. Alexich
Vice President
Nuclear Engineering
American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43215

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Mr. Wade Schuler, Supervisor
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W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
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U.S. Nuclear Regulatory Commission
Resident Inspectors Office
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Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Honorable Jim Catania, Mayor
City of Bridgman, Michigan 49106

U.S. Environmental Protection Agency
Region V Office
ATTN: EIS COORDINATOR
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Maurice S. Reizen, M.D.
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The Honorable Tom Corcoran
United States House of Representatives
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