MAR 8 1978

Docket No. 50-316

Indiana & Michigan Electric Company
Indiana & Michigan Power Company
ATTN: Mr. John Tillinghast
Vice President
P. O. Box 18
Bowling Green Station
New York, New York 10004

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 2 - DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

The Commission has issued the enclosed Amendment No. 2 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. This amendment authorizes Mode 2 operation. In addition, as a result of our audit of environmental qualification records, we have specified in this amendment four new conditions requiring additional documentation prior to March 22, 1978.

Copies of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,
Original Signed By
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 2 to License No. DPR-74
2. Safety Evaluation
3. Notice of Issuance

ccs w/encls:
See page 2

Const. 1
GD

OFFICE➤	LWR-2:LPM	ELD	LWR-2:BC	LWR:AD	DPM:DD	DPM:D
SURNAME➤	MMlynczak:ld	S TREBY	KKniel	DVassallo	RDeYoung	RBoyd
DATE➤	03/ 8 /78	03/ 8 /78	03/ 8 /78	03/ 8 /78	03/ 8 /78	03/ 8 /78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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Distribution - AMENDMENT NO. 2 TO DPR-74 (D. C. COOK, UNIT 2), DTD. 3/8/78

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JYore
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

INDIANA AND MICHIGAN ELECTRIC COMPANY
INDIANA AND MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The issuance of this amendment is in compliance with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-74 is amended to authorize Mode 2 operation and delete the following license conditions:

Paragraph 2.C(3)(b)	Steam Generator Subcompartment Pressure Response Analysis
Paragraph 2.C(3)(f)	Electrical Connectors
Paragraph 2.C(3)(p)	Emergency Planning
Paragraph 2.C(3)(q)	Qualification of Electrical Equipment

3. The conditions specified in Paragraph D. of Attachment No. 1 to License No. DPR-74 have also been resolved as follows:

   A. Items 1 through 5 of paragraph D have been resolved and are deleted.

B. Item 6 of paragraph D has been superceded by the following condition:

Prior to Mode 1 operation not to exceed 678 megawatts thermal (twenty percent of rated power) resolution of the following conditions is required. Written approval by the Commission is required prior to operation at greater than twenty percent of rated power.

1. Indiana and Michigan Power Company shall, within ninety days of the date of this amendment, provide for staff review the results of qualification testing in conformance with IEEE 323-1971 requirements, to environmentally qualify the Foxboro E11GM, Foxboro E13DM and Barton 764 transmitters in safety-related circuits inside containment for a postulated steamline break. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, provide for staff review justification for continued operation of D. C. Cook Unit 2 until the qualification testing program has been completed.

2. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, March 22, 1978, provide for staff review documentation of steamline break qualification test procedures and results for all electrical terminations in all safety-related systems inside containment.

3. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, March 22, 1978, provide for staff review documentation of the test procedures and results used to qualify Continental instrument cable for steamline break. Indiana and Michigan Power Company shall, in addition, provide within two weeks of the date of this amendment, March 22, 1978, the results of a comparability analysis to demonstrate the environmental qualification of Continental cable currently installed.

4. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, March 22, 1978, provide for staff review documentation of the comparability of electrical penetrations installed in D. C. Cook Unit 2 to those prototype penetrations tested under steamline break environmental conditions.

4. Facility Operating License No. DPR-74 is further amended by the addition of the following:

   Page 9, last paragraph (regarding physical security plan) - insert the letter "D." prior to the text of the paragraph.

5. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 8, 1978

INDIANA AND MICHIGAN ELECTRIC COMPANY
INDIANA AND MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. DPR-74

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   A. The issuance of this amendment is in compliance with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-74 is amended to authorize Mode 2 operation and delete the following license conditions:

Paragraph 2.C(3)(b)	Steam Generator Subcompartment Pressure Response Analysis
Paragraph 2.C(3)(f)	Electrical Connectors
Paragraph 2.C(3)(p)	Emergency Planning
Paragraph 2.C(3)(q)	Qualification of Electrical Equipment

3. The conditions specified in Paragraph D. of Attachment No. 1 to License No. DPR-74 have also been resolved as follows:

   A. Items 1 through 5 of paragraph D have been resolved and are deleted.

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B. Item 6 of paragraph D has been superceded by the following condition:

Prior to Mode 1 operation not to exceed 678 megawatts thermal (twenty percent of rated power) resolution of the following conditions is required. Written approval by the Commission is required prior to operation at greater than twenty percent of rated power.

1. Indiana and Michigan Power Company shall, within ninety days of the date of this amendment, provide for staff review the results of qualification testing in conformance with IEEE 323-1971 requirements, to environmentally qualify the Foxboro E11GM, Foxboro E13DM and Barton 764 transmitters in safety-related circuits inside containment for a postulated steamline break. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, provide for staff review justification for continued operation of D. C. Cook Unit 2 until the qualification testing program has been completed.

2. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, March 22, 1978, provide for staff review documentation of steamline break qualification test procedures and results for all electrical terminations in all safety-related systems inside containment.

3. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, March 22, 1978, provide for staff review documentation of the test procedures and results used to qualify Continental instrument cable for steamline break. Indiana and Michigan Power Company shall, in addition, provide within two weeks of the date of this amendment, March 22, 1978, the results of a comparability analysis to demonstrate the environmental qualification of Continental cable currently installed.

4. Indiana and Michigan Power Company shall, within two weeks of the date of this amendment, March 22, 1978, provide for staff review documentation of the comparability of electrical penetrations installed in D. C. Cook Unit 2 to those prototype penetrations tested under steamline break environmental conditions.

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SURNAME➤						
DATE➤						

4. Facility Operating License No. DPR-74 is further amended by the addition of the following:

   Page 9, last paragraph (regarding physical security plan) - insert the letter "D." prior to the text of the paragraph.

5. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 8, 1978

OFFICE	DPM:LWR 2	DPM:LWR 2	ELD	DPM:LWR	DPM:DD	DPM:D
SURNAME	Lee/ MMlynczak:ld	KKniel	S TREBY	DBVassallo	RDeYoung	RSBoyd
DATE	3/8/78	3/8/78	3/8/78	3/8/78	3/9/78	3/9/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

AMENDMENT 2 TO DPR-74

INDIANA AND MICHIGAN POWER COMPANY
INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT 2

DOCKET NO. 50-316

This safety evaluation presents NRC staff acceptance of documentation and analyses supporting the amendment of Facility Operating License No. DPR-74 for D. C. Cook Unit 2. This amendment involves deletion of conditions described in paragraphs 2.C(3)(b) and 2.C(3)(p) of DPR-74; the basis for these deletions is staff review and approval of documentation supporting the revised steam generator subcompartment pressure response analysis and the licensees' emergency plan. This amendment also deletes condition 2.C(3)(q) on the basis of staff review and interim approval of the licensees' presentation of information regarding environmental qualification of certain electrical equipment inside containment for steamline break. Each of these items is discussed in this safety evaluation. In addition, the Office of Inspection and Enforcement has verified that the licensees have satisfactorily completed the requirements of paragraph 2.C(3)(f) of DPR-74 and of paragraph D of the Attachment No. 1 to DPR-74.

1. Steam Generator Subcompartment Pressure Response

(a) Steam Generator Subcompartment Analysis

In Supplement 7 to the Safety Evaluation Report for the Donald C. Cook Nuclear Plant Unit 2 dated December 23, 1977, we reported on our review of the licensees' preliminary analysis of the steam generator enclosure subcompartment analysis. We concluded that the licensees' provided sufficient information to indicate that analytical methods and approaches used are consistent with those which we previously reviewed and found acceptable on a similar plant, and based on the analysis provided, we did not expect that modifications to the steam generator enclosure or supports would be required. We further stated that this was subject to confirmation prior to power operation. Facility Operating License No. DPR-74 was therefore conditioned to require the licensees to finalize the steam generator compartment analysis and submit it for Commission review and approval prior to Mode 2 operation of D. C. Cook Unit 2.

By a letter dated January 23, 1978 to Mr. Case from Mr. Tillinghast of the Indiana and Michigan Power Company, the licensees' submitted the finalized steam generator subcompartment analysis. Supplemental information describing the subcompartment nodalization study was submitted in a letter from Mr. Maloney of the Indiana and Michigan Power Company to Mr. Case dated February 27, 1978.

We have reviewed the above information and we conclude that the licensees' steam generator subcompartment methods of analysis, modeling and pressure time history results are acceptable for evaluation of both the steam generator enclosure structures and the steam generator supports.

As stated in Supplement 7 to the Safety Evaluation Report for the D. C. Cook Nuclear Plant, Unit 2 the licensees considered the analysis submitted in November of 1977 to be preliminary because 1) the results had not been subjected to their quality assurance verification program and 2) they had not completed the nodalization sensitivity study. In finalizing the steam generator subcompartment analyses, the pressure time history calculated by the licensees has not changed from the values which were reported in November 1977 and discussed in Supplement 7 to the Safety Evaluation Report.

In completing our review, we have not performed confirmatory calculations of a steam generator enclosure transient response. As stated in Supplement 7 to the Safety Evaluation Report, we have previously reviewed the computer code (TMD code) used by the licensees as a part of the Commission's topical report evaluation program and concluded that the TMD code is an acceptable code for the evaluation of subcompartment transient response. Furthermore, we performed a confirmatory analysis for the McGuire Nuclear Station steam generator enclosure; the steam generator enclosures for the McGuire Station are similar to those in the Donald C. Cook Nuclear Plant Units 1 and 2 and were analyzed by the licensees using the TMD code and the same nodalization arrangement used for the D. C. Cook Nuclear Plant steam generator subcompartment analysis. As we reported in our safety evaluation for the McGuire Nuclear Station, we found reasonable agreement between the results of our confirmatory analysis and the licensees' analysis. We, therefore, felt it unnecessary to perform confirmatory calculations for the D. C. Cook Nuclear Plant steam generator enclosures.

In keeping with the requirements of standard review plan 6.2.1.2, "Subcompartment Analysis," we reviewed the licensees' input assumptions and parameters used in their analysis of the steam generator enclosure transient response and the nodalization sensitivity study.

In addition to the nine node model used to generate the structural loads on the steam generator enclosure and supports, the licensees also performed subcompartment analyses using five node and a seventeen node representations of the free volume within the steam generator enclosures. In going from the nine node model to a seventeen node model, the peak horizontal force predicted to act on the steam generator was reduced by about 20% (i.e., from about 1,000,000 pounds to about 8,000,000 pounds) while the peak negative moment acting upon the steam generator was increased by about 50% (i.e., from about 4,600,000 ft-lbs. to about 6,850,000 ft-lbs). The peak moment and peak horizontal force occur at different times during the steam generator enclosure transient response. The analysis of the steam generator supports capability was made by combining the force and moments determined from the nine node model without consideration of the time phasing. The licensees have stated that this conservative combination of the loads is more severe than the combination of loads from the seventeen node model using a time phase analysis.

As stated above, based upon our review, we find the licensees' steam generator subcompartment analysis to be acceptable for evaluation of the adequacy of both the steam generator enclosure structures and the steam generator supports.

(b) Steam Generator Subcompartment Structural Adequacy

An assessment of the structural integrity of the steam generator enclosure has been made by the licensees. On January 23, 1978, the licensees submitted to NRC the "Steam Generator Subcompartment Pressure Response Analysis" report. The structural aspects of the analysis are contained in Attachment No. 2 to the report, which described the structural analysis methods and the computer programs used and the acceptance criteria. Various aspects of the analysis were also provided by the licensees.

The licensees utilized the FELAP computer program to perform the analysis. FELAP is a general purpose computer program used for the analysis of three dimensional elastic structures composed of shells, plates and straight or curved beams. The program computes the dynamic and static response to distributed, thermal and concentrated loads, including the spatially varying and dynamically applied transient pressure loading. The steam generator compartment was modeled as a series of quadrilateral finite elements with constant modulus of elasticity (E) and constant plate moment of inertia (D). The dynamic transient pressure loads acting simultaneously with the design basis earthquake were considered to act together with the operating load conditions. The results from the program are the joint deflections, mid-panel stresses and stress resultants and joint moments and forces (shear and in plane).

These moments, axial forces and shears were compared with the ultimate capacities of the structural section at points of interest as a measure of the factor of safety against the loading combination involving the compartment pressurization. The licensees have established a factor of safety of 1.5 as the acceptance criterion to prevent steam leakage into the upper containment volume bypassing the ice condenser. The staff concurs with this objective.

On the basis of the information provided by the licensees, we have made an evaluation of the structural adequacy of the steam generator subcompartment. The use of a factor of safety of 1.5 for loading combinations involving such an unlikely event is reasonably conservative. For the portions of the subcompartment which could constitute a bypass between the steam generator subcompartment and the upper volume, the lowest factor of safety as computed by the licensees is 1.48. Even though this is less than 1.50, we find such a deviation is acceptable considering the fact that (1) the occurrence of such an event is highly improbable; (2) the factor of safety of 1.50 is conservative; (3) the deviation occurs for a localized region; (4) the actual strength of the wall section is most likely to be higher than that used in the calculation; and (5) the licensees verified the structural adequacy of the steam generator subcompartment using the ACI-318, 1971 Code regarding the allowable shear stress on sections subjected to significant tension. On the basis of the above considerations, we conclude that the structural design of the steam generator subcompartment is reasonably conservative and will not adversely affect the bypass capability of the ice condenser.

(c) Steam Generator Supports

In Supplement 7 to the Safety Evaluation Report, we reported on our review of the licensees' preliminary analysis of the steam generator supports. We concluded that the licensees have provided sufficient information to indicate that no modifications of the support structure would be required. The operating license was, however, conditioned to require the licensees to finalize the steam generator compartment analysis and submit for Commission review and approval prior to Mode 2 operation of the plant.

By a letter dated January 23, 1978 to Mr. Case from Mr. Tillinghast of Indiana and Michigan Company, the licensees submitted the results of the finalized steam generator subcompartment analysis and in Attachment 3 addressed the steam generator supports analysis. We reviewed the above information and concluded that the method of analysis of the steam generator supports was acceptable, but the criteria including any limitations on component buckling used for determining the design load capacity of the vertical column support system were not provided. The licensees have since then provided this information and based on our review of the same, we conclude that the design of the steam generator supports is acceptable.

During a postulated main steam line break at the side of the steam generator and the Design Basis Earthquake, the critical element of the upper supports is the belly band. As the steam generator is supported by the belly band through one-way acting bumpers, the band will carry applied loading in tension and bending. The design criteria for the band was that the combined stress due to bending and tension produced by the design load must not exceed the material yield stress.

The critical load for the lower steam generator vertical supports is compressive under the combination of normal loads, DBE, pipe breaks and steam generator compartment pressurization. The AISC-69 equations were used to establish allowable buckling load for the faulted conditions by applying a multiplication factor of 1.67 to the AISC equation to permit an increase in the normal design condition allowable AISC load for the low probability postulated pipe break event. With this applied load factor stresses in the support are limited to below the material yield strength. The calculated stress in the vertical columns was found to be 41.3 ksi versus an allowable of 43.5 ksi.

As stated above, based on our review, the design of the steam generator supports is found to be acceptable.

2. Documentation of Qualification Test Procedures and Results for Safety-Related Electrical Equipment for a Postulated Steamline Break

The licensees have provided and the staff has reviewed documentation of the qualification test procedures and results used to verify the ability of specific safety-related electrical equipment to function in the environment existing inside containment after a postulated steamline break. The staff reviewed the qualification documentation for the following equipment: Foxboro E11GM, Foxboro E13DM and Barton 764 transmitters, Rosemount 176 KF and KS detectors, Cerro and Continental instrument cable, and containment electrical penetrations. In addition, the staff requested qualification documentation for all terminations in safety-related circuits inside containment. Based on our review of documentation presented, we have concluded that existing qualification test procedures and results are adequate to satisfy condition 2.C(3)(q) of Facility Operating License No. DPR-74. However, we have developed requests for additional environmental qualification testing and documentation of existing test results prior to authorizing operation of the facility at greater than twenty percent of rated power. Our evaluations and requirements for additional testing and documentation follow for each type of equipment reviewed.

(a) Instrumentation: Foxboro E11GM, Foxboro E13DM and Barton 764 Transmitters

The licensees have presented and the staff has reviewed documentation of qualification tests for Foxboro E11GM, Foxboro E13DM and Barton 764 transmitters. The licensees have referenced documentation provided on the North Anna review for identical instruments. On the basis of this reference to test information already provided, we have issued conditional approval of the licensees documentation. Our approval is conditional upon the licensees commitment to provide within two weeks, the formal documentation of qualification information presented to the staff in support of qualification of these transmitters and within ninety days the results of a complete environmental qualification testing program for these transmitters in conformance with IEEE 323-1971. Test conditions must include sequential radiation exposure, seismic testing and exposure to environmental conditions expected following a postulated steamline break and/or loss-of-coolant accident, whichever is most limiting.

Our approval is also conditioned upon the licensees' commitment to provide within two weeks a basis for continued operation of the facility until all required environmental qualification testing has been completed.

(b) Cerro, Anaconda and Continental Cable

We have evaluated the licensees documentation of test procedures and results for environmental qualification of Cerro instrument cable for operation in a postulated steamline break environment. We have found the licensees' documentation acceptable on the basis of the equivalence of the Cerro instrument cable to Cerro control cable which was prototype tested.

We have reviewed the licensees' documentation of the test procedures and results for environmental qualification of Anaconda power cable for operation in a postulated steamline break environment. We have found the licensees' documentation acceptable based on their description of the test profile and results presented.

We have reviewed the licensees' qualification test results for tests of the materials used in Continental instrument cable. Based on our evaluation of results of these materials tests, we have conditionally approved the licensees' documentation. We require that the licensee document within two weeks all test procedures and results used to qualify Continental cable.

(c) Penetrations

The licensees have provided results of irradiation tests performed on electrical penetrations. In addition the licensees have provided for staff review the results of Conax medium voltage penetration qualification tests which were performed during the D. C. Cook Unit 1 splice tests in November 1977. Based on this information, we have approved on a conditional basis the licensees' qualification test procedures and results. Our approval is conditioned upon the licensees' commitment to provide, within two weeks, documentation in the form of a comparability study which demonstrates the equivalence of penetrations tested to those installed at the facility.

(d) Rosemount 176 KF and KS Detectors

The licensees have presented, and the staff has reviewed, documentation of qualification tests for Rosemount 176 KF and KS detectors. The licensees have referenced documentation provided on the North Anna review for identical detectors. This information satisfied the Commission requirements and was found acceptable. Therefore the D. C. Cook Unit 2 Rosemount 176 KF and KS detectors also are found acceptable.

(e) Terminations in Safety-Related Circuits

We have required that the licensee provide within two weeks documentation of all test procedures used and results obtained in qualification testing of all terminations in all safety-related circuit inside containment. All terminations should be qualified to steamline break profile.

3. Emergency Plan

On December 21, 1977, the licensees submitted the revised emergency plan for the D. C. Cook Nuclear Plant. The revised plan was filed as Amendment 80 to the Final Safety Analysis Report and documents all revisions to the plan proposed previously by the licensees.

On the basis of this filing, we find that the licensees have fulfilled their obligation regarding documentation of revisions to the D. C. Cook Nuclear Plant emergency plan and have satisfied the requirements stated in our regulations. Therefore, the condition stated in Paragraph 2.C(3)(p) of Facility Operating License No. DPR-74 has been satisfied and may be deleted.

4. Electrical Connectors

Our Office of Inspection and Enforcement has informed us that all ITT-Cannon electrical connectors in the lower containment have been replaced with qualified splices as stated in license condition 2.C(3)(f) and in paragraph D, item 2 of Attachment 1 to Facility Operating License No. DPR-74.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

M. M. Mlynczak

M. M. Mlynczak, Project Manager
Light Water Reactors Branch No. 2
Division of Project Management

Karl Kniel

Karl Kniel, Chief
Light Water Reactors Branch No. 2
Division of Project Management

Dated: March 8, 1978

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

AMENDMENT 2 TO DPR-74

INDIANA AND MICHIGAN POWER COMPANY
INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT 2

DOCKET NO. 50-316

This safety evaluation presents NRC staff acceptance of documentation and analyses supporting the amendment of Facility Operating License No. DPR-74 for D. C. Cook Unit 2. This amendment involves deletion of conditions described in paragraphs 2.C(3)(b) and 2.C(3)(p) of DPR-74; the basis for these deletions is staff review and approval of documentation supporting the revised steam generator subcompartment pressure response analysis and the licensees' emergency plan. This amendment also deletes condition 2.C(3)(q) on the basis of staff review and interim approval of the licensees' presentation of information regarding environmental qualification of certain electrical equipment inside containment for steamline break. Each of these items is discussed in this safety evaluation. In addition, the Office of Inspection and Enforcement has verified that the licensees have satisfactorily completed the requirements of paragraph 2.C(3)(f) of DPR-74 and of paragraph D of the Attachment No. 1 to DPR-74.

1. Steam Generator Subcompartment Pressure Response

   (a) Steam Generator Subcompartment Analysis

   In Supplement 7 to the Safety Evaluation Report for the Donald C. Cook Nuclear Plant Unit 2 dated December 23, 1977, we reported on our review of the licensees' preliminary analysis of the steam generator enclosure subcompartment analysis. We concluded that the licensees' provided sufficient information to indicate that analytical methods and approaches used are consistent with those which we previously reviewed and found acceptable on a similar plant, and based on the analysis provided, we did not expect that modifications to the steam generator enclosure or supports would

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be required. We further stated that this was subject to confirmation prior to power operation. Facility Operating License No. DPR-74 was therefore conditioned to require the licensees to finalize the steam generator compartment analysis and submit it for Commission review and approval prior to Mode 2 operation of D. C. Cook Unit 2.

By a letter dated January 23, 1978 to Mr. Case from Mr. Tillinghast of the Indiana and Michigan Power Company, the licensees' submitted the finalized steam generator subcompartment analysis. Supplemental information describing the subcompartment nodalization study was submitted in a letter from Mr. Maloney of the Indiana and Michigan Power Company to Mr. Case dated February 27, 1978.

We have reviewed the above information and we conclude that the licensees' steam generator subcompartment methods of analysis, modeling and pressure time history results are acceptable for evaluation of both the steam generator enclosure structures and the steam generator supports.

As stated in Supplement 7 to the Safety Evaluation Report for the D. C. Cook Nuclear Plant, Unit 2 the licensees considered the analysis submitted in November of 1977 to be preliminary because 1) the results had not been subjected to their quality assurance verification program and 2) they had not completed the nodalization sensitivity study. In finalizing the steam generator subcompartment analyses, the pressure time history calculated by the licensees has not changed from the values which were reported in November 1977 and discussed in Supplement 7 to the Safety Evaluation Report.

In completing our review, we have not performed confirmatory calculations of a steam generator enclosure transient response. As stated in Supplement 7 to the Safety Evaluation Report, we have previously reviewed the computer code (TMD code) used by the licensees as a part of the Commission's topical report evaluation program and concluded that the TMD code is an

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acceptable code for the evaluation of subcompartment transient response. Furthermore, we performed a confirmatory analysis for the McGuire Nuclear Station steam generator enclosure; the steam generator enclosures for the McGuire Station are similar to those in the Donald C. Cook Nuclear Plant Units 1 and 2 and were analyzed by the licensees using the TMD code and the same nodalization arrangement used for the D. C. Cook Nuclear Plant steam generator subcompartment analysis. As we reported in our safety evaluation for the McGuire Nuclear Station, we found reasonable agreement between the results of our confirmatory analysis and the licensees' analysis. We, therefore, felt it unnecessary to perform confirmatory calculations for the D. C. Cook Nuclear Plant steam generator enclosures.

In keeping with the requirements of standard review plan 6.2.1.2, "Subcompartment Analysis," we reviewed the licensees' input assumptions and parameters used in their analysis of the steam generator enclosure transient response and the nodalization sensitivity study.

In addition to the nine node model used to generate the structural loads on the steam generator enclosure and supports, the licensees also performed subcompartment analyses using five node and a seventeen node representations of the free volume within the steam generator enclosures. In going from the nine node model to a seventeen node model, the peak horizontal force predicted to act on the steam generator was reduced by about 20% (i.e., from about 1,000,000 pounds to about 8,000,000 pounds) while the peak negative moment acting upon the steam generator was increased by about 50% (i.e., from about 4,600,000 ft-lbs. to about 6,850,000 ft-lbs). The peak moment and peak horizontal force occur at different times during the steam generator enclosure transient response. The analysis of the steam generator supports capability was made by combining the force and moments determined from the nine node model without consideration of the time phasing. The licensees have stated that this conservative combination of the loads is more severe than the combination of loads from the seventeen node model using a time phase analysis.

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As stated above, based upon our review, we find the licensees' steam generator subcompartment analysis to be acceptable for evaluation of the adequacy of both the steam generator enclosure structures and the steam generator supports.

(b) Steam Generator Subcompartment Structural Adequacy

An assessment of the structural integrity of the steam generator enclosure has been made by the licensees. On January 23, 1978, the licensees submitted to NRC the "Steam Generator Subcompartment Pressure Response Analysis" report. The structural aspects of the analysis are contained in Attachment No. 2 to the report, which described the structural analysis methods and the computer programs used and the acceptance criteria. Various aspects of the analysis were also provided by the licensees.

The licensees utilized the FELAP computer program to perform the analysis. FELAP is a general purpose computer program used for the analysis of three dimensional elastic structures composed of shells, plates and straight or curved beams. The program computes the dynamic and static response to distributed, thermal and concentrated loads, including the spatially varying and dynamically applied transient pressure loading. The steam generator compartment was modeled as a series of quadrilateral finite elements with constant modulus of elasticity (E) and constant plate moment of inertia (D). The dynamic transient pressure loads acting simultaneously with the design basis earthquake were considered to act together with the operating load conditions. The results from the program are the joint deflections, mid-panel stresses and stress resultants and joint moments and forces (shear and in plane).

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These moments, axial forces and shears were compared with the ultimate capacities of the structural section at points of interest as a measure of the factor of safety against the loading combination involving the compartment pressurization. The licensees have established a factor of safety of 1.5 as the acceptance criterion to prevent steam leakage into the upper containment volume bypassing the ice condenser. The staff concurs with this objective.

On the basis of the information provided by the licensees, we have made an evaluation of the structural adequacy of the steam generator subcompartment. The use of a factor of safety of 1.5 for loading combinations involving such an unlikely event is reasonably conservative. For the portions of the subcompartment which could constitute a bypass between the steam generator subcompartment and the upper volume, the lowest factor of safety as computed by the licensees is 1.48. Even though this is less than 1.50, we find such a deviation is acceptable considering the fact that (1) the occurrence of such an event is highly improbable; (2) the factor of safety of 1.50 is conservative; (3) the deviation occurs for a localized region; (4) the actual strength of the wall section is most likely to be higher than that used in the calculation; and (5) the licensees verified the structural adequacy of the steam generator subcompartment using the ACI-318, 1971 Code regarding the allowable shear stress on sections subjected to significant tension. On the basis of the above considerations, we conclude that the structural design of the steam generator subcompartment is reasonably conservative and will not adversely affect the bypass capability of the ice condenser.

(c) Steam Generator Supports

In Supplement 7 to the Safety Evaluation Report, we reported on our review of the licensees' preliminary analysis of the steam generator supports. We concluded that the licensees have provided sufficient information to indicate that no modifications of the support structure would be required. The operating license was, however, conditioned to require the licensees to finalize the steam generator compartment analysis and submit for Commission review and approval prior to Mode 2 operation of the plant.

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By a letter dated January 23, 1978 to Mr. Case from Mr. Tillinghast of Indiana and Michigan Company, the licensees submitted the results of the finalized steam generator subcompartment analysis and in Attachment 3 addressed the steam generator supports analysis. We reviewed the above information and concluded that the method of analysis of the steam generator supports was acceptable, but the criteria including any limitations on component buckling used for determining the design load capacity of the vertical column support system were not provided. The licensees have since then provided this information and based on our review of the same, we conclude that the design of the steam generator supports is acceptable.

During a postulated main steam line break at the side of the steam generator and the Design Basis Earthquake, the critical element of the upper supports is the belly band. As the steam generator is supported by the belly band through one-way acting bumpers, the band will carry applied loading in tension and bending. The design criteria for the band was that the combined stress due to bending and tension produced by the design load must not exceed the material yield stress.

The critical load for the lower steam generator vertical supports is compressive under the combination of normal loads, DBE, pipe breaks and steam generator compartment pressurization. The AISC-69 equations were used to establish allowable buckling load for the faulted conditions by applying a multiplication factor of 1.67 to the AISC equation to permit an increase in the normal design condition allowable AISC load for the low probability postulated pipe break event. With this applied load factor stresses in the support are limited to below the material yield strength. The calculated stress in the vertical columns was found to be 41.3 ksi versus an allowable of 43.5 ksi.

As stated above, based on our review, the design of the steam generator supports is found to be acceptable.

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2. Documentation of Qualification Test Procedures and Results for Safety-Related Electrical Equipment for a Postulated Steamline Break

The licensees have provided and the staff has reviewed documentation of the qualification test procedures and results used to verify the ability of specific safety-related electrical equipment to function in the environment existing inside containment after a postulated steamline break. The staff reviewed the qualification documentation for the following equipment: Foxboro E11GM, Foxboro E13DM and Barton 764 transmitters, Rosemount 176 KF and KS detectors, Cerro and Continental instrument cable, and containment electrical penetrations. In addition, the staff requested qualification documentation for all terminations in safety-related circuits inside containment. Based on our review of documentation presented, we have concluded that existing qualification test procedures and results are adequate to satisfy condition 2.C(3)(q) of Facility Operating License No. DPR-74. However, we have developed requests for additional environmental qualification testing and documentation of existing test results prior to authorizing operation of the facility at greater than twenty percent of rated power. Our evaluations and requirements for additional testing and documentation follow for each type of equipment reviewed.

(a) Instrumentation: Foxboro E11GM, Foxboro E13DM and Barton 764 Transmitters

The licensees have presented and the staff has reviewed documentation of qualification tests for Foxboro E11GM, Foxboro E13DM and Barton 764 transmitters. The licensees have referenced documentation provided on the North Anna review for identical instruments. On the basis of this reference to test information already provided, we have issued conditional approval of the licensees documentation. Our approval is conditional upon the licensees commitment to provide within two weeks, the formal documentation of qualification information presented to the staff in support of qualification of these transmitters and within ninety days the results of a complete environmental qualification testing program for these transmitters in conformance with IEEE 323-1971. Test conditions must include sequential radiation exposure, seismic testing and exposure to environmental conditions expected following a postulated steamline break and/or loss-of-coolant accident, whichever is most limiting.

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Our approval is also conditioned upon the licensees' commitment to provide within two weeks a basis for continued operation of the facility until all required environmental qualification testing has been completed.

(b) Cerro, Anaconda and Continental Cable

We have evaluated the licensees documentation of test procedures and results for environmental qualification of Cerro instrument cable for operation in a postulated steamline break environment. We have found the licensees' documentation acceptable on the basis of the equivalence of the Cerro instrument cable to Cerro control cable which was prototype tested.

We have reviewed the licensees' documentation of the test procedures and results for environmental qualification of Anaconda power cable for operation in a postulated steamline break environment. We have found the licensees' documentation acceptable based on their description of the test profile and results presented.

We have reviewed the licensees' qualification test results for tests of the materials used in Continental instrument cable. Based on our evaluation of results of these materials tests, we have conditionally approved the licensees' documentation. We require that the licensee document within two weeks all test procedures and results used to qualify Continental cable.

(c) Penetrations

The licensees have provided results of irradiation tests performed on electrical penetrations. In addition the licensees have provided for staff review the results of Conax medium voltage penetration qualification tests which were performed during the D. C. Cook Unit 1 splice tests in November 1977. Based on this information, we have approved on a conditional basis the licensees' qualification test procedures and results. Our approval is conditioned upon the licensees' commitment to provide, within two weeks, documentation in the form of a comparability study which demonstrates the equivalence of penetrations tested to those installed at the facility.

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(d) Rosemount 176 KF and KS Detectors

The licensees have presented, and the staff has reviewed, documentation of qualification tests for Rosemount 176 KF and KS detectors. The licensees have referenced documentation provided on the North Anna review for identical detectors. This information satisfied the Commission requirements and was found acceptable. Therefore the D. C. Cook Unit 2 Rosemount 176 KF and KS detectors also are found acceptable.

(e) Terminations in Safety-Related Circuits

We have required that the licensee provide within two weeks documentation of all test procedures used and results obtained in qualification testing of all terminations in all safety-related circuit inside containment. All terminations should be qualified to steamline break profile.

3. Emergency Plan

On December 21, 1977, the licensees submitted the revised emergency plan for the D. C. Cook Nuclear Plant. The revised plan was filed as Amendment 80 to the Final Safety Analysis Report and documents all revisions to the plan proposed previously by the licensees.

On the basis of this filing, we find that the licensees have fulfilled their obligation regarding documentation of revisions to the D. C. Cook Nuclear Plant emergency plan and have satisfied the requirements stated in our regulations. Therefore, the condition stated in Paragraph 2.C(3)(p) of Facility Operating License No. DPR-74 has been satisfied and may be deleted.

4. Electrical Connectors

Our Office of Inspection and Enforcement has informed us that all ITT-Cannon electrical connectors in the lower containment have been replaced with qualified splices as stated in license condition 2.C(3)(f) and in paragraph D, item 2 of Attachment 1 to Facility Operating License No. DPR-74.

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Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Original signed by

M. M. Mlynczak, Project Manager
Light Water Reactors Branch No. 2
Division of Project Management

Original signed by
K. Kniel

Karl Kniel, Chief
Light Water Reactors Branch No. 2
Division of Project Management

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY
INDIANA AND MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to Facility Operating License No. DPR-74, issued to Indiana and Michigan Electric Company and Indiana and Michigan Power Company, which authorizes Mode 2 operation for the Donald C. Cook Nuclear Plant, Unit No. 2 (the facility) located in Berrien County, Michigan. The amendment is effective as of its date of issuance. This action is a part of the licensing action encompassed in the "Notice of Consideration of Issuance of Facility Operating Licenses and Notice of Opportunity for Hearing Pursuant to 10 CFR Part 50, Appendix D, Section C."

Facility Operating License No. DPR-74 and its Attachment No. 1 contained several conditions requiring staff approval prior to initial criticality. These items have been completed to the satisfaction of the Commission and the appropriate restrictions have been removed in Amendment No. 2. Amendment 2 requires resolution of four additional conditions relating to qualification of electrical equipment inside containment prior to operation of the facility in Mode 1 at greater than twenty percent of rated power.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the

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Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Part 51.5 (d) (4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) Amendment No. 2 to License No. DPR-74, and (2) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Maude Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan. A copy of items (1) and (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland, this 8th day of March, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
K. Kniel

Karl Kniel, Chief
Light Water Reactors Branch No. 2
Division of Project Management

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