October 24, 2001

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB0739 AND MB0740)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 256 to Facility Operating License No. DPR-58 and Amendment No. 239 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments are in response to your application dated October 24, 2000, as supplemented June 29, 2001, approve changes to the updated final safety analysis report (UFSAR).

The amendments would approve changes to the UFSAR to incorporate a supplemental methodology for the analysis of steam generator overfill following a steam generator tube rupture.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 256 to DPR-58
2. Amendment No. 239 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

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DATE	10/2/01	9/28/01	8/17/2001	10/19/01	10/23/01

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Donald C. Cook Nuclear Plant, Units 1 and 2

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 256
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 24, 2000, as supplemented June 29, 2001, complies 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 application, 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, by Amendment No. 256, Facility Operating License No. DPR-58 is hereby amended to authorize a change to the methodology used in the evaluation of a steam generator (SG) overfill following a SG tube rupture referenced in the updated final safety analysis report (UFSAR), as set forth in the license amendment application dated October 24, 2000, as supplemented June 29, 2001, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William D. Reckley, Acting Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 24, 2001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 239
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 24, 2000, as supplemented June 29, 2001, complies 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 application, 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, by Amendment No. 239, Facility Operating License No. DPR-74 is hereby amended to authorize a change to the methodology used in the evaluation of a steam generator (SG) overfill following a SG tube rupture referenced in the updated final safety analysis report (UFSAR), as set forth in the license amendment application dated October 24, 2000, as supplemented June 29, 2001, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William D. Reckley, Acting Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 24, 2001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 256 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 239 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated October 24, 2000, as supplemented June 29, 2001, the Indiana Michigan Power Company (the licensee) requested approval of changes to the updated final safety analysis report (UFSAR). The proposed amendments would approve incorporating a supplemental methodology into the analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The new analysis more realistically models the operator actions and plant response. The results from the calculations would then be used to determine the time available for operator actions to prevent overfill and to revise the plant Emergency Operating Procedures.

The current SGTR analysis for D. C. Cook Nuclear Plant (CNP) Units 1 and 2, calculates an average break flow from the time of the accident until the reactor trips and safety injection (SI) initiates. Following SI initiation, the UFSAR analysis uses an equilibrium break flow that continues at a constant rate for 30 minutes. The resulting break flow mass transfer is then used to calculate the radiological consequences of the SGTR. The UFSAR calculations are considered conservative since the reduction in the break flow rate over the thirty minute period is ignored. Inherent in this evaluation is the assumption that the operator can terminate the break flow in 30 minutes, and that the termination will prevent the SG from overfilling. The 30 minute time to termination is the current licensing basis referenced in the UFSAR.

During an expanded system readiness review, the licensee conducted SGTR simulator exercises, and the operators demonstrated that the time to terminate the tube rupture break flow exceeded the 30 minute assumption. Should operators significantly exceed the 30 minute termination criteria, the SG could overfill, assuming the break flow documented in the UFSAR occurred. Upon SG overfill, water enters the steam lines. The steam lines at D. C. Cook are not qualified to contain water, and the pressure operated relief valves (PORVS) are not qualified to relieve water. This event could have radiological consequences beyond the design basis of the D. C. Cook Nuclear Plants.

The licensee proposes to incorporate a supplemental methodology into the analysis of steam generator overfill following a SGTR. The licensee plans to use the results from the analysis to determine the time available for operator actions to prevent overfill and to revise the plant Emergency Operating Procedures.

The licensee' s June 29, 2001, letter, provided additional information only in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information dated May 7, 2001. The letter provided clarifying information within the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The discrepancy between the simulator exercises and the assumptions in the SGTR analysis, required the licensee to reexamine the SGTR accidents. The licensee proposes implementing WCAP-10698-P-A "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," to update the analysis. To implement the WCAP, the licensee used the LOFTTR2 computer code and the plant-specific current licensing basis assumptions. The improvements in the analysis included modeling operator actions, enhancing the SG secondary side model, and improving the tube rupture break flow model.

From the new analysis, the licensee plans to determine the time available for operator actions to prevent steam generator overfill. They will then correct the discrepancy between the UFSAR analysis and the demonstrated plant response with regard to SGTR overfill. Based upon the results of the calculations, the licensee also plans to update the emergency operating procedures to incorporate the enhancements identified by the new analysis.

The licensee's use of WCAP-10698-P-A presented several exceptions from the requirements of the NRC staff's safety evaluation that approved it. These exceptions are presented in more detail below as Items 1 through 5, and are summarized in Table 1. Overall, WCAP-10698-P-A describes a conservative methodology for SGTR overfill calculations. The WCAP applies the LOFTTR1 computer code for modeling the SGTR transient. Input assumptions into the accident analysis model include loss of offsite power, most reactive rod stuck, conservative initial conditions, turbine runback, 120 percent of 1971 ANS (American Nuclear Society) decay heat standard, and the worst single failure, among other assumptions. However, variations in plant designs prevent a single model from adequately representing all Westinghouse plants. Because of these variations, the NRC required that plant-specific input be provided when utilities reference the WCAP methodology. The required plant-specific input is as follows:

1. Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design-basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention, using design-basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

The licensee confirmed that they have simulator and training programs in place that assure the necessary actions and times can be taken to prevent steam generator overfill. The licensee also stated that the operator action times assumed in the analysis are realistic because they were confirmed by simulator demonstrations. They incorporated these action times into their training programs. However, when establishing the bases for the SGTR transient, the licensee did not assume the same design-basis assumptions as the WCAP-10698 analysis.

2. A site specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in Reference (1) - (Note: Reference (1) of the WCAP-10698 Safety Evaluation).

In response to this requirement, the licensee did not submit the offsite consequence analysis. They, instead, proposed keeping their current licensing basis SGTR calculation for the offsite radiological consequences. The licensee states that this proposal is adequate, since the original licensing basis analysis has a greater primary to secondary side mass transfer than the analysis performed using the WCAP methodology.

3. An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.

Because licensee's evaluation determined that the steam generators do not overfill, the licensee did not submit a structural evaluation of the main steam lines and associated supports for the water-filled conditions of a SGTR overfill incident.

4. A list of systems, components and instrumentation which are credited for accident mitigation in the plant-specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves, specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components, state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design-basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify SG sampling as a means of ruptured SG identification, provide the expected time period for obtaining the sample results and discuss the effects on the duration of the accident.

The licensee provided the required lists of systems, components, and instrumentation that are used for SGTR accident mitigation. They also specified the safety classification of the systems and power sources. However, the licensee listed several systems used for SGTR mitigation that are not safety related and do not have safety related backups. The licensee justified the use of the non-safety related equipment by stating that these systems are credited in the current UFSAR Section 14.2.4 accident analysis. Upon review of UFSAR Section 14.2.4, the staff concludes that the licensing basis SGTR analysis does credit limited use of non-safety grade equipment for mitigating the SGTR.

5. A survey of plant primary and "balance-of-plant" systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.

The licensee did not use the bounding plant analysis of WCAP-10698-P-A, but performed a D. C. Cook plant-specific analysis. Therefore, they did not submit a survey of the plant systems that determined the variance with the WCAP analysis. Also, the licensee did not assume the worst single failure as prescribed by the WCAP-10698-P-A safety analysis, and did not provide it's effect on the margin to overfill. The licensee based their decision not to assume the worst single failure on the fact that their current licensing basis does not include a single failure.

Overall, the proposed D. C. Cook plant methodology and the approved WCAP-10698 methodology vary significantly. The differences are listed in Table 1.

Table 1

WCAP-10698-P-A	**D. C. Cook**
Uses LOFTTR1 computer model	Uses LOFTTR2 computer model
Assumes a worst single failure for design basis	Does not assume worst single failure
Requires offsite radiological consequence analysis, assuming worst failure	Does not provide analysis based on WCAP methodology or worst failure
Requires structural adequacy analysis of SGTR overfill	Does not provide structural analysis
Requires comparison of BOP equipment with WCAP case	Does not provide comparison

Because of the significance of these variations from the WCAP-10698-P-A methodology, the staff cannot endorse the licensee's reference to WCAP-10698-P-A in the licensing basis for a SGTR overfill analysis. However, the LOFTTR2 computer code has been previously accepted for licensing basis SGTR analyses. Therefore, we find it acceptable for the licensee to use LOFTTR2 for their licensing basis SGTR overfill analysis as well as the licensing calculation for the design basis SGTR accident.

Finally, the licensee committed to revise the Unit 1 emergency operating procedures to incorporate enhancements from the new analysis. The licensee has updated the Unit 1and 2 emergency operating procedures. The licensee will revise the UFSAR to incorporate the use of the new methodology in the SGTR analyses. Incorporating the information from the new SGTR model is acceptable. However, as stated earlier, citing WCAP-10698-P-A as the licensing basis for the SGTR overfill analysis is not acceptable.

3.0 SUMMARY

The licensee's method of implementation of WCAP-10698-P-A does not satisfy the WCAP safety evaluation requirements. Therefore, the incorporation of WCAP-10698-P-A into D. C. Cook 's licensing basis is not acceptable as proposed in the October 24, 2000, application. However, the LOFTTR2 computer code has been previously accepted for use in licensing basis analyses. Therefore, the NRC staff concludes that the licensee can incorporate the LOFTTR2 code into its licensing bases for CNP and can use the LOFTTR2 code, with the current licensing basis assumptions as inputs for the overfill analysis of steam generator tube rupture accidents. This change to the licensing basis does not affect accident initiators or precursors. The change also does not increase the probability of an accident or decrease the ability of the operators to mitigate the consequences of an accident. Therefore, the proposed change to the UFSAR is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (66 FR 7682). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Sean Peters

Date: October 24, 2001