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INDIANA MICHIGAN POWER

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C1000-11
10 CFR 50.90

Docket Nos.: 50-315
50-316

U.S. Nuclear Regulatory Commission
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Mail Stop O-P1-17
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Donald C. Cook Nuclear Plant Units 1 and 2
LICENSE AMENDMENT FOR CHANGES IN STEAM
GENERATOR TUBE RUPTURE ANALYSIS METHODOLOGY

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company, the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Facility Operating Licenses DPR-58 and DPR-74 to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of CNP to a postulated SGTR with respect to CNP SG overfill. The break flow model used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. Since this methodology is not currently part of the CNP licensing basis, use of this methodology for the supplemental SGTR analysis has been determined to involve an unreviewed safety question. Therefore, NRC staff review and approval are required for the addition of this methodology to the CNP licensing basis in accordance with 10 CFR 50.90. The Updated Final Safety Analysis Report will be revised to incorporate the use of the new methodology.

Attachment 1 to this letter provides a detailed description of the proposed changes. Attachment 2 describes the evaluation performed in accordance with 10 CFR 50.92(c), which concludes that no significant hazard is involved.

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Attachment 3 provides the environmental assessment and Attachment 4 provides the new commitments made in this letter.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. Wayne J. Kropp, Director of Regulatory Affairs, at (616) 697-5056.

Sincerely,

R. P. Powers
Vice President

\dmb

Attachments

c: J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

I, Robert P. Powers, being duly sworn, state that I am Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

R. P. Powers
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 24th DAY OF October, 2000

Julie Newmiller
Notary Public

My Commission Expires 8-22-04

JULIE E. NEWMILLER
Notary Public, Berrien County, MI
My Commission Expires Aug 22, 2004

ATTACHMENT 1 TO C1000-11

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGES

A. Summary of Proposed Changes

Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Facility Operating Licenses DPR-58 and DPR-74 to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company (Westinghouse) WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of CNP to a postulated SGTR with respect to CNP SG overfill. The break flow model used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report (SER), "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. The use of this methodology for the supplemental SGTR analysis has been determined to involve an unreviewed safety question (USQ). Therefore, NRC staff review and approval are required for the addition of this methodology to the CNP licensing basis in accordance with 10 CFR 50.90. The Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the use of the new methodology.

The change in methodology would be applied to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected SG in response to an SGTR event. The present methodology would be retained for calculating the radiological consequences of a postulated SGTR since the current CNP licensing basis methodology continues to bound the radiological consequences calculated by the new methodology.

B. Description and Bases for the Current Requirements

The CNP licensing basis analysis for SGTR was included in the December 18, 1967, license application. This analysis was retained as part of the Final Safety Analysis Report, Section 14.2.4, "Steam Generator Tube Rupture," as described in the NRC's SER dated September 10, 1973, and remains in Section 14.2.4 of the UFSAR. This analysis calculates an average SGTR break flow for the time period from the postulated accident initiation until reactor trip and safety injection (SI) initiation, which are assumed to occur simultaneously. This analysis assumes an equilibrium break flow following reactor trip and SI initiation that continues at a constant rate for thirty minutes. The resulting break flow mass transfer is then used to calculate the radiological consequences of the postulated accident.

The calculations performed as part of the SGTR analysis methodology are based on the limited industry experience and calculational capabilities that existed when CNP was licensed. The use of a constant mass flow rate for thirty minutes simplified the analysis calculations and was deemed conservative because the reduction in the break flow rate over the thirty-minute duration is ignored. Inherent in the evaluation is the assumption that an operator can terminate the break flow in thirty minutes, and that a thirty-minute termination time will prevent SG overfill from occurring.

C. Description of the Change and the Need for Revision of the Requirement

I&M proposes to adopt as part of the CNP licensing basis a new, NRC-approved analysis method using a different calculational technique that models operator actions to more accurately predict the time to overfill an SG following an SGTR. The current licensing basis analysis for the postulated CNP SGTR accident assumes that the SG experiencing the tube rupture could be isolated within thirty minutes. That estimate was considered reasonable based on what was known when CNP Units 1 and 2 were licensed. Since that time, additional information has been obtained, procedures have been changed, control room protocols have evolved, and better analysis tools have become available for calculating operational transient responses.

Issues related to the SGTR analysis and the time to SG overfill were identified in March 1999, during the CNP expanded system readiness review. To address these issues, I&M personnel conducted SGTR simulator exercises that adhere to the CNP emergency operating procedures (EOPs). These exercises demonstrated that the time for reactor operators to terminate the tube rupture break flow in the affected SG exceeds thirty minutes. Therefore, the analysis assumption that the affected SG is isolated within thirty minutes is invalid.

The methodology used for the analysis of SGTR events cannot be used to determine if the SG overfills since the simplistic modeling does not consider the secondary side liquid volume transient. The methodology uses a mass and energy balance to calculate the primary-to-secondary leakage through the ruptured tube and steam release to the atmosphere. At thirty minutes, the event is assumed to end as a result of the assumption that primary-to-secondary break flow has been terminated. No operator actions are explicitly modeled in the analysis, although the operators are required to perform the necessary steps to terminate the break flow. Thirty minutes is not assumed a criterion for operator action; it is an analysis assumption used to conservatively calculate the total break flow.

Although the thirty-minute timeframe was selected as an analytical assumption, it could result in water entering the main steam line connected to the affected SG. Overfilling of the affected SG is unacceptable because the steam lines are not qualified for water fill and the SG power-operated relief valves are not qualified to relieve water. Overfill during a SGTR may propagate the event into a main steam line break accident coincident with SGTR. The radiological consequences

from such a combination of events have not been analyzed as it is outside of the licensing and design basis for CNP.

I&M determined that a new analysis using a different calculational technique that includes modeling operator actions is needed to more accurately identify the time to overfill. I&M proposes to use the break flow model and associated LOFTRR2 computer code described in WCAP-10698-P-A to quantify SG filling response following a postulated SGTR. The use of this break flow model would supplement the present SGTR analysis that did not specifically address SG overfill.

I&M has performed calculations in accordance with the WCAP-10698-P-A methodology. The new analysis calculation, which accounts for timing of operator actions, indicates that the break flow duration is longer than thirty minutes. The incorporation of a new methodology to more accurately characterize SGTRs and subsequent SG overfill timing into the CNP licensing basis is needed to rectify this non-conformance with the UFSAR description of the CNP response to SGTR events. Incorporation of the new methodology into the CNP licensing basis has been determined to constitute a USQ, requiring review and approval by the NRC.

D. Bases for the Proposed Changes

The current SGTR thermal and hydraulic analyses performed for CNP do not incorporate the advantage of the available computer analysis model to determine the dynamic plant behavior following an SGTR accident. Rather, simplified calculations were performed based on the expected SGTR transient response to determine the primary to secondary break flow and the steam release to the atmosphere for use in calculating the offsite radiation exposure due to the event.

Following the R. E. Ginna Nuclear Power Plant SGTR event in January 1982, a new SGTR analysis methodology was developed. The new methodology is based on modeling operator actions for SGTR recovery derived from plant-specific simulator studies, incorporates an improved SG secondary-side model, and includes a more realistic tube rupture break flow model. The NRC approved the revised SGTR analysis methodology in an SER, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. As required by the NRC as part of this SER, this methodology has subsequently been adopted by the Westinghouse Owners Group subgroup of plants that received operating licenses in this time frame. I&M has determined that the use of this methodology is applicable to CNP.

The current CNP SGTR analysis does not explicitly address the SG filling response due to an SGTR accident. The new methodology, widely adopted by owners of Westinghouse pressurized water reactor plants, is best suited to analyze the SG secondary-side inventory response following an SGTR as opposed to simplified hand calculations. Thus, approval for use of the

methodology is being requested to supplement the CNP SGTR analysis. This will allow the SGTR analysis to reflect more accurately the time to overfill of the affected SG in the CNP licensing basis.

The use of this WCAP methodology and associated computer code for break flow modeling more accurately calculates the plant response to an SGTR event. The improved accuracy of the new methodology provides valuable information related to the analysis of operator actions and the associated timing of these actions. The more accurate calculations related to the SGTR overfill transient provide valuable insights for enhancement of the EOPs. These types of improvements have been incorporated into the Unit 2 EOPs and are in progress for the EOPs for Unit 1. Revisions to the appropriate Unit 1 EOPs to incorporate enhancements identified due to the new SGTR SG overfill methodology will be completed prior to restart from the current outage.

E. Discussion of Risk

The proposed change, to adopt a new analytical method to evaluate the effects of an SGTR, does not affect any accident initiators or precursors. As such, the proposed change does not increase the probability of an accident. The proposed change also does not affect the ability of operators to mitigate the consequences of an accident. The WCAP-10698-P-A methodology calculates more accurately the flow from the reactor coolant system to the SG secondary side following a postulated SGTR. The resulting break flow mass transfer is then used to calculate the radiological consequences of the postulated accident. The current licensing basis methodology for calculating the radiological consequences of a postulated SGTR results in mass transfer values that bound those calculated using the new methodology. As such, the existing licensing basis radiological consequence calculations will be retained. Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the UFSAR will not be increased. Therefore, the incorporation of the new methodology would not result in an increase in risk for CNP.

ATTACHMENT 2 TO C1000-11

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Indiana Michigan Power Company (I&M) has evaluated this proposed amendment and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license does not involve a significant hazard if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously evaluated; or

3. involve a significant reduction in a margin of safety.

I&M proposes to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of Donald C. Cook Nuclear Plant (CNP) to a postulated SGTR with respect to CNP SG overfill. The break flow model used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. The use of this methodology for the supplemental SGTR analysis has been determined to involve an unreviewed safety question. Therefore, NRC staff review and approval are required for the addition of this methodology to the CNP licensing basis in accordance with 10 CFR 50.90. The Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the use of the new methodology.

The change in methodology would be applied to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected SG in response to an SGTR event. The present methodology would be retained for calculating the radiological consequences of a postulated SGTR since the current CNP licensing basis methodology continues to bound the radiological consequences calculated by the new methodology.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed change, to adopt a new analytical method to evaluate the effects of an SGTR, does not affect any accident initiators or precursors. As such, the proposed change does not increase the probability of an accident. The proposed change also does not affect the ability of operators to mitigate the consequences of an accident. The proposed change does not impact the design of the affected plant systems such that previously analyzed systems, structures, and components (SSCs) would now be more likely to fail. The changes will not modify plant systems to reduce their design capability during normal operating and accident conditions. The use of the WCAP-10698-P-A methodology to more accurately calculate the flow from the reactor coolant system (RCS) to the SG secondary side following a postulated SGTR does not affect the probability of any analyzed events. The use of the WCAP-10698-P-A methodology does not affect SGTR initiators or precursors. Therefore, incorporating the new methodology does not affect equipment malfunction probability, nor does it affect or create new accident initiators or precursors. Thus, there will be no reduction in the capability of those SSCs in limiting the consequences of previously evaluated accidents.

Additionally, the present methodology for calculating the radiological consequences of a postulated SGTR is conservative when compared with results from the new methodology. As such, the existing licensing basis radiological consequence calculations will be retained. Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the UFSAR will not be increased. The use of this WCAP methodology and associated computer code for break flow modeling more accurately calculates the plant response to an SGTR event. The improved accuracy of the new methodology provides valuable information related to the analysis of operator actions and the associated timing. Such accurate transient response information enables enhancements to be made to the emergency operating procedures (EOPs).

Therefore, the proposed changes cannot increase the consequences or probability of occurrence of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not impact the design of affected plant systems, involve a physical alteration to the systems, or change to the way in which systems are currently operated, such that previously unanalyzed SGTRs would now occur. The change to incorporate the WCAP-10698-P-A methodology does not introduce any new malfunctions; it calculates more accurately the flow from the RCS to the SG secondary side following a postulated SGTR to determine the time available for operator actions to prevent overfilling the affected SG.

Thus, use of the WCAP-10698-P-A methodology does not affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

Therefore, the proposed changes do not increase the possibility of a new or different kind of accident than previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The approval of the license amendment will not result in any modifications to affected plant systems that would reduce their design capabilities during normal operating and accident conditions. By using the WCAP-10698-P-A methodology, a more accurate SGTR response is calculated. The improved understanding of the transient response enables enhancements to the EOPs, which provide further assurance that SSCs required for accident mitigation are protected.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that these changes involve no significant hazards consideration.

ATTACHMENT 3 TO C1000-11

ENVIRONMENTAL ASSESSMENT

Indiana Michigan Power Company (I&M) has evaluated this license amendment request (LAR) against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this LAR meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment 2, this proposed amendment does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The change in methodology would be applied to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected steam generator in response to a steam generator tube rupture (SGTR) event. The present methodology would be retained for calculating the radiological consequences of a postulated SGTR since the current Donald C. Cook Nuclear Plant licensing basis methodology continues to bound the radiological consequences calculated by the new methodology. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in significant changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT 4 TO C1000-11

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this submittal. Other actions discussed in the submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
Revisions to the appropriate Unit 1 emergency operating procedures to incorporate enhancements identified due to the new steam generator tube rupture overfill methodology will be completed prior to restart from the current outage.	Prior to initial entry into Mode 2 from the current Unit 1 outage.
After NRC approval of the amendment request, the Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the use of the new methodology in steam generator tube rupture analyses.	To be included in the next 10 CFR 50.71(e) UFSAR update submittal.