November 30, 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. MB1975 AND MB1976)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 260 to Facility Operating License No. DPR-58 and Amendment No. 243 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The amendments consist of changes to the Technical Specifications (T/S) in response to your application dated May 17, 2001, as supplemented by letters dated September 5, 2001, and November 29, 2001.

The amendments would revise T/S 3/4.9.3, "Decay Time," to allow the start of core offload at 100 hours after reactor subcriticality between September 15 and June 15, when the ultimate heat sink temperature is assumed to be not higher than 77.8 °F, and 148 hours after reactor subcriticality between June 16 and September 14, when the ultimate heat sink temperature is assumed to be not higher than 85 °F. T/S 3/4.9.3 currently prohibits fuel movement in the reactor pressure vessel until the reactor has been subcritical for at least 168 hours. The 168-hour decay time was placed in the CNP T/S with Amendment Nos. 169 and 152 to DPR-58 and DPR-74, respectively, on January 14, 1993.

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,

/S. Miranda for/

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. 260 to DPR-58
2. Amendment No. 243 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

Mr. Robert P. Powers, Senior Vice President November 30, 2001
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Docket Nos. 50-315 and 50-316
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cc w/encls: See next page
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AVegel, RIII	ACRS	JStang	JHannon	

*See previous concurrence
Accession No. **ML012910720** * Provided SE input by memo

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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. 260
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 May 17, 2001, as supplemented by letter dated September 5, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 260, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/D. Hood for/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: November 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 260

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 9-3	3/4 9-3
B 3/4 9-1	B 3/4 9-1
	B 3/4 9-1a

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 243
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 May 17, 2001, as supplemented by letter dated September 5, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 243, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/D. Hood for/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: November 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 243

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 9-3	3/4 9-3
B 3/4 9-1	B 3/4 9-1
	B 3/4 9-1a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 260 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 243 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 letter dated May 17, 2001, as supplemented by letters dated September 5, 2001 and November 29, 2001, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, requested an amendment to Appendix A, Technical Specifications (T/S), of Facility Operating Licenses DPR-58 and DPR-74, which would revise T/S 3/4.9.3, "Decay Time," to allow the start of core offload at 100 hours after reactor subcriticality between September 15 and June 15, when the ultimate heat sink temperature is assumed to be not higher than 77.8 °F, and 148 hours after reactor subcriticality between June 16 and September 14, when the ultimate heat sink temperature is assumed to be not higher than 85 °F. T/S 3/4.9.3 currently prohibits fuel movement in the reactor pressure vessel until the reactor has been subcritical for at least 168 hours. The 168-hour decay time was placed in the CNP T/S with Amendment Nos. 169 and 152 to DPR-58 and DPR-74, respectively, on January 14, 1993.

Delaying movement of recently irradiated fuel allows time for radioactive decay of the fission product inventory in the fuel, and thereby reduces the amount of decay heat that must be removed by the spent fuel pool (SFP) cooling systems. I&M has requested a reduction in the required decay time because they expect that decay time could become a critical path item during future refueling outages.

A reduction in decay time can lead to an increase in peak SFP water and concrete temperatures, due to a relatively higher level of decay heat generation in the most recent batch of offloaded fuel. A reduction in decay time can also affect the radiological results of the fuel handling accident (FHA) analysis. I&M has determined that the 168-hour decay time can be reduced without adversely impacting safety.

Accordingly, implementation of this license amendment request is considered in terms of its potential effects upon heat removal from the SFP water, heating of the SFP concrete, and the CNP FHA analysis.

The supplemental letter of September 5, 2001, contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

2.0 EVALUATION

2.1 Fuel Handing Accident (FHA)

When the decay time requirement was increased from 100 to 168 hours, on January 14, 1993, the decay time that was assumed in the FHA analysis was retained at 100 hours.

The radiological consequences of an FHA were previously analyzed in the CNP updated final safety analysis report (UFSAR). There are two analyses, one for an FHA in the fuel handling building, and one for an FHA in the containment. The FHA in the fuel handling building analysis was performed in support of the SFP re-rack approved in Amendment Nos. 169 and 152. The FHA in the containment analysis was performed in support of revisions to T/S 3.9.4, in Amendment Nos. 197 and 182, to allow both containment personnel airlocks to be open during core alterations. Both of these analyses assumed a decay period of only 100 hours.

With regard to the radiation doses to control room personnel, I&M re-analyzed the CNP FHA, to account for the selective implementation of the alternative source term (AST) to control room habitability analyses. The reanalysis of the control room doses, due to an FHA, also assumed a decay time of 100 hours. The analysis was performed in support of I&M's June 12, 2000, license amendment request for control room habitability and Generic Letter 99-02 requirements. This license amendment request proposed using the AST in design-basis control room habitability assessments. The staff has found the AST analysis of the FHA acceptable, and has issued License Amendment Nos. 258 and 241 to account for the selective implementation of the AST to control room habitability analyses for the CNP FHA.

The staff found that there is reasonable assurance that radiological consequences of an FHA at CNP would result in doses that meet the acceptance criteria of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC-19, as clarified in NUREG-0800 Sections 6.4 and 15.7.4.

Since all the CNP FHA analyses of interest performed since 1993 have been based upon a decay time assumption of 100 hours, and since I&M's request, to change T/S 3/4.9.3 to allow a 100-hour decay time between September 15 and June 15 and a 148-hour decay time between June 16 and September 14 does not involve the application of any decay times less than 100 hours, the CNP FHA analysis results would not be affected by the proposed TS revision.

2.2 Decay Heat Generation

2.2.1 Background

NUREG-0800, "Standard Review Plan", Reference 2, provides criteria related to the design and performance of the spent fuel pool. Section 9.2.5, "Ultimate Heat Sink," includes Branch Technical Position (BTP) ASB 9-2, which describes a method for calculating residual decay energy in light-water reactors. The BTP method is based upon American Nuclear Society 5.1, "Decay Heat Power for Light Water Reactors" (October 1979). The licensee's decay heat generation rates were calculated using a combination of the BTP method and ORIGEN2, obtained from the Oak Ridge National Laboratory's Computer Code Collection. The BTP method can be used for time dependent calculations, whereas ORIGEN2 cannot. The BTP method and ORIGEN2 were applied to three benchmark cases, and, as expected, the BTP

method yielded the more conservative (higher) results. Adjustment factors were determined from comparisons of the benchmark case results, and used to adjust the BTP method results, in subsequent cases, to produce results that would be expected to be produced by ORIGEN2, if ORIGEN2 were applicable to time dependent analyses. The licensee's hybrid methodology is not evaluated or approved, since the licensee's submittal does not contain the detail needed to perform such an evaluation. This safety evaluation is limited to an evaluation of the licensee's results.

2.2.2 Decay Heat Generation Rate

The licensee evaluated three scenarios, Cases 1a, 1b, and 2, based upon the cases discussed in NUREG-0800, "Standard Review Plan," (SRP) Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." Cases 1a and 1b are planned heat load cases with differences in decay time and ultimate heat sink temperature assumptions. Case 2 is an unplanned heat load case, since it reflects refueling of adjacent units with just 30 days' separation, rather than the intended separation of several months. Case 2 is therefore, a conservative evaluation of the SFP temperature transient, during refueling with both SFP cooling trains available.

CASE 1a:
A full-core discharge of 193 assemblies after 148 hours of decay time is considered. The previously discharged refueling load of 88 assemblies was assumed to have been discharged five months earlier. The single-failure considered is the loss of one train of SFP cooling. The design-basis ultimate heat sink temperature of 85 °F is assumed.

CASE 1b:
Case 1a with a 100 hour decay time and a reduced ultimate heat sink temperature of 77.8 °F.

CASE 2:
A full-core discharge of 193 assemblies after 100 hours of decay time is considered. The previously discharged refueling load of 88 assemblies was assumed to have been discharged 30 days earlier. The design basis ultimate heat sink temperature of 85 °F is assumed. Both SFP cooling trains are assumed to be lost at a time when the peak SFP temperature exists so as to minimize the time to boil the water in the SFP.

ALL CASES:
Decay heat consists of decay heat from the last full core offload, decay heat from the last permanent discharge, and long-term decay heat from all previously discharged fuel assemblies, plus all projected fuel assembly offloads. The last full core offload takes 38.6 hours to insert all 193 fuel assemblies into the SFP, causing the total decay heat to rise until the offloading is completed.

To determine the bounding cases for maximum decay heat calculations, the licensee made the following conservative assumptions:

1. All reactor thermal power levels are increased by 2 percent to account for the plant's reactor thermal power calorimetric uncertainty.

2. Expected bounding parameters (i.e., unit power rating, burnup, batch size, etc.) are used for all projected discharges.

3. The total fuel inventory stored in the SFP equals the full capacity of the SFP (3613 assemblies).

4. With the exception of the last two refueling loads, a value of 1223 effective full power days (EFPD) is assumed for future discharges. This is based on a 97 percent capacity factor for an 18-month (549-day) cycle length resulting in 532.5 EFPD per cycle.

The Nuclear Regulatory Commission (NRC) staff agrees with the conservative assumptions the licensee used to calculate the decay heat loads.

The licensee's benchmark methodology, based upon a hybrid of ORIGEN2 and BTP calculations, is not evaluated here, due to a lack of supporting information. Rather, the licensee's results have been compared with the results of the staff's independent calculations, which were based upon the BTP methods. These comparisons indicate that the licensee's results are reasonable.

The last full core offload contributes the greatest part of the total decay heat value, and this is the contribution that is most sensitive to the post-criticality hold time. The licensee's and staff's calculations for this portion of the decay heat load were in reasonable agreement.

As expected, the long-term decay heat from all previously discharged fuel assemblies, including all projected fuel assembly offloads, is a constant value; and not dependent upon case definition. The licensee's ORIGEN2 calculation of this value yields 11.18 MBTU/hr, which, according to the licensee's benchmark method, should correspond to 81.12 percent of the BTP value. Application of the licensee's 81.12 percent benchmark adjustment to the staff's BTP result yields 11.05 MBTU/hr. In an alternative approach, the BTP's specification of 16,000 hours of operating history is applied for all past and future offloads, and the resulting long-term decay heat value, 11.17 MBTU/hr, is very close to the licensee's value, without the use of any adjustment factors.

The licensee's total decay heat generation rate was compared to the NRC staff's BTP calculation results for each time step listed in the licensee's submittal. For comparison purposes, the licensee's higher (more conservative) value for long-term decay heat (11.18) was assumed in the overall BTP decay heat transient calculations. Except for a two-hour period in the beginning of the Case 2 transient, when the BTP results are slightly higher, the licensee's decay heat generation rates consistently exceed the staff's BTP predictions.

2.2.3 Spent Fuel Pool Cooling System Evaluation

CNP Units 1 and 2 share a common SFP. The SFP cooling system transfers decay heat from the water in the SFP to the component cooling water system, and then to the essential service water system. The SFP cooling system has two parallel trains, each with a pump and heat

exchanger. Each pump has a design flow rate of 2300 gpm and each train has a design heat transfer rate of 14.9 million BTU/hr. One train is associated with each unit.

(from CNP UFSAR, Table 9.4-2)

Spent Fuel Cooling System	Shell	Tube
Design pressure (psig)	150	150
Design temperature (°F)	200	200
Design flow rate (million lb/hr)	1.49	1.14
Design inlet temperature (°F)	95	120
Design outlet temperature (°F)	105	106.9

In evaluating the maximum SFP bulk temperature, the licensee made the following conservative assumptions.

1. There are no heat losses from the SFP water to concrete or air, and no evaporative heat transfer is assumed.

2. The thermal performance of the SFP cooling system heat exchangers is determined with all heat transfer surfaces fouled to their design-basis maximum levels.

There were no plugged tubes assumed for the SFP cooling system heat exchangers. This assumption was justified by the licensee, based upon a problem-free operating history, with respect to the heat exchanger tubes.

For all cases, a full core discharge was assumed, in order to bound partial discharge scenarios. Therefore, the licensee's analysis and the staff's safety evaluation both consider the "planned" discharge to implement a full core offloading. For Cases 1a and 1b, the maximum pool bulk temperature for the planned full-discharge scenario was calculated assuming only one SFP cooling system pump and one SFP cooling system heat exchanger were available, which is consistent with the single active failure recommendation of the SRP guidelines. Case 2, the unplanned full-core discharge scenario, assumed that both SFP cooling system pumps and heat exchangers were available until the SFP water temperature reached its peak. Then all SFP cooling was assumed to be lost. Then the SFP water temperature was allowed to rise to the boiling point, in order to determine the minimum time for the SFP water to reach the boiling point. For Cases 1a and 1b, the licensee's results end at 60 hours, when SFP temperature peaks. For Case 2, their results end at 54 hours, when the SFP water begins to boil. The following table shows the maximum pool bulk temperature calculated for each scenario:

Discharge Scenario	Peak Pool Bulk Temperature (°F)
Case 1a: planned offload, 148 hr delay; 85 °F ultimate heat sink temp; one SFP cooling train fails	179.9
Case 1b: planned offload, 100 hr delay; 77.8 °F ultimate heat sink temp; one SFP cooling train fails	180.0
Case 2: unplanned offload, 100 hr delay; 85 °F ultimate heat sink temp; no failure until peak SFP temp is reached - then all cooling is lost, leading to boiling in the SFP	142.3 (5.8 hrs to boiling)
Case 2: Case 2 above; but noting the time to boiling after pool temperature reaches 180 °F	180.0 (2.7 hrs to boiling)

As shown in the above table, the maximum bulk temperature in the SFP water does not exceed 180 °F for either of the planned offload scenarios.

Case 2 addresses the complete loss of cooling scenario, in which the SFP water temperature begins to rise and eventually reaches the boiling temperature. The licensee performed an analysis to determine the minimum time from the loss-of-pool cooling at peak pool water temperature to the time the pool begins to boil, based on the heat load for the full core offload. The analysis results indicate that there would be about 5.8 hours available for corrective actions prior to SFP boiling in the unlikely event of a total failure of forced cooling to the SFP. The time remaining to boiling, once the pool temperature reaches 180°F is 2.7 hrs. The maximum boiloff rate is expected to be less than 120 gpm. There is at least 500 gpm of make up flow available.

Both planned offload cases, presented by the licensee, predict that the SFP temperature could reach 180 °F, assuming that one of the two trains of SFP cooling fails. The staff recognizes that certain maintenance activities affecting SFP cooling support systems may be performed with the fuel removed from the reactor vessel, to provide a higher overall level of safety. However, with one SFP cooling train removed from service, the remaining cooling train may not provide adequate reliability of the SFP cooling function, considering the relatively higher decay heat generation rate and the consequential reduction in available recovery time. To improve the overall reliability of SFP cooling, the licensee, by letter dated November 29, 2001, has committed to revise their shutdown safety and risk management procedure to ensure that both SFP cooling trains will remain available from initiation of core offload to completion of core reload, with exceptions covered by specific limitations and contingencies (Reference 5). Specifically, the licensee has committed to:

> Ensure that one SFP cooling train is not made unavailable during the time from initiation of core offload to completion of core reload unless contingencies exist to readily restore the train to service before pool design temperature is reached.
>
> and
>
> Minimize the time of unavailability of a SFP cooling train during the time from initiation of core offload to completion of core reload.

2.2.4 Conclusion

For planned refueling conditions with both cooling trains in service, SFP water temperature will stay below 142.3°F, which is below the long term SFP design temperature of 150°F. In the event that one SFP cooling train fails, the remaining SFP cooling train will maintain SFP temperature below the SFP design temperature of 180°F. The licensee has committed to implement administrative controls to ensure an adequate likelihood that one train of SFP cooling will be in operation, or that cooling capability will be recovered prior to the SFP water temperature exceeding 180°F.

In the unlikely event of a sustained loss of both SFP cooling trains, the available makeup capacity exceeds the maximum potential rate of evaporative losses, and these makeup sources can be aligned within the time available prior to the onset of boiling. Therefore, the staff concludes that the reliability and capacity of SFP cooling and makeup systems are adequate to deal with the increased heat load, resulting from the proposed reduction in decay time, and are consistent with the capabilities specified in GDC 61.

2.3 Heating of the SFP Concrete

The staff has reviewed I&M's submittal of May 17, 2001, from a structural standpoint. I&M requested approval to use American Concrete Institute (ACI) 349-97 to establish acceptability of the peak SFP temperature with respect to the concrete integrity, since ACI-318-63, which is cited in the CNP UFSAR, does not provide any limitations pertaining to maximum short-term concrete temperature. Accordingly, the staff has considered the applicability of the ACI code used in the SFP structural evaluation.

The NRC staff concurs with the licensee's conclusion that a peak SFP temperature that is above 150 °F, but not exceeding 180 °F, for about 33 days under the worst case scenario of core offload will not adversely affect the integrity of the concrete in the SFP. Therefore, ACI-349-97 may be used to establish acceptability of the peak SFP temperature with respect to the concrete integrity, to support the requested revision of TS 3/4.9.3, "Decay Heat Time."

2.3.1 Background

The SFP structural evaluation, to support revision of TS 3/4.9.3, "Decay Heat Time," is based upon American Concrete Institute ACI-349-97, "Code Requirements for Nuclear Safety Related Concrete Structures." ACI-318-63, "Building Code Requirements for Reinforced Concrete Structures," which is described in Section 5.2, "Containment Structure," of the CNP updated final safety analysis report UFSAR, does not provide any limitations with respect to maximum short-term concrete temperature. Therefore, I&M is requesting approval to use ACI-349-97 to establish acceptability of the peak SFP temperature with respect to the concrete integrity. In connection with the requested revision of the TS, the staff's structural review is centered upon the appropriateness of the ACI code used in the SFP structural evaluation.

2.3.2 Evaluation

As described in Reference 1, I&M performed a time-dependent analysis of decay heat loads and the SFP water heat-up. I&M found that the results of this analysis showed that the available train of SFP cooling provides sufficient cooling to limit the SFP bulk temperature to 180 °F under the worst case scenario: full core offload with a failure of one train of SFP cooling. Since the design-basis SFP bulk pool temperature is 150°F, I&M examined the integrity of the concrete, liner and associated structural elements under the increased SFP bulk pool temperature, and found that they can withstand up to a peak SFP temperature of 185 °F (which bounds the 180 °F peak SFP cooling analysis temperature). I&M found that the leak-tightness and integrity of the liner, concrete, and associated structural elements will be maintained during the predicted peak SFP temperature scenario (Reference 1).

Whereas ACI-318-63, which is referenced in the CNP UFSAR, does not provide any limitations with respect to maximum short-term concrete temperature, ACI-349-97 Appendix A, "Thermal Considerations," has established two temperature limits. The first limit, which is for normal operation or any other long-term exposure, specifies that the concrete temperature shall not exceed 150°F except for local areas such as around penetrations, where temperature increases not to exceed 200°F are allowed. The second temperature limitation is for an accident or any other short-term exposure. The limit is that the temperature shall not exceed 350 °F for the surface. I&M states that the 350 °F limit would apply to the SFP concrete walls and floor when they are exposed to elevated temperatures, such as during the end of fuel pool full core discharge, including cooling single-failure and high ultimate heat sink water temperature (Reference 1). I&M states that it is reasonable to apply the ACI -49-97 to this short-term temperature loading as an unusual event. This position is consistent with that discussed in a memorandum regarding resolution of spent fuel storage pool action plan issues from J. M. Taylor (NRC) to Chairman Jackson, et al., dated July 26, 1996. Section 3.2.1, "Structural Considerations," of the Spent Fuel Pool Action Plan Status Report discusses the application of ACI 349-97 when evaluating concrete structures such as the SFP. This report states that, "...during a rise in the SFP bulk temperature due to temporary loss of forced cooling, the low thermal diffusivity of concrete and the large thermal capacity of the SFP concrete cause the temperature distribution within the concrete structure to change slowly after a rise in the temperature. Evaporative cooling of the pool limits the maximum temperature attainable at the concrete surface following a temporary loss of forced cooling". I&M's SFP concrete element calculations demonstrated structural integrity during the peak SFP water temperature event. Further, I&M's calculation for the temperature effect on the SFP liner showed that under the increased temperature condition the liner maintains its leak tightness for the load combinations to which it is subjected.

In response to an NRC staff request for additional information to justify that its consideration of the number of days it takes for the peak temperature to come down to the licensing basis temperature of 150 °F can be classified as "short-term," I&M stated that, based on the worst-case scenario (148 hour decay time), the maximum duration that the water temperature remains above 150 °F is about 788 hours (33 days) with a peak temperature of 180 °F, Reference 3. I&M further states that ACI-349-97 does not provide explicit definition of "short-term," nor does there appear to be any widely recognized time period to which "short-term" is typically applied throughout the industry. Section A.4.1 of the code allows prolonged exposure of up to 200 °F around penetrations. Section A.4.2 allows temperature excursions of up to

350 °F for accidents or any other short-term period and up to 650 °F from steam or water jets in the event of a pipe failure. Citing the Commentary section of ACI-349-97, I&M contends that the linking of the temperature limits of Section A.4.2 to "accidents" implies that the mission time of accident response systems can be taken as the definition of "short-term," and that the duration of the transient can be considered "short-term" as well.

In response to the August 10, 2001, request for additional information, I&M provided available test data to show that the reduced concrete strength due to the increased temperature during the worst-case conditions for the spent fuel pool will not cause deterioration of the concrete either with or without load, as required by ACI code 349-97, I&M provided certain data from available technical literature to support its conclusion that a temperature of 180 °F will have minimal effects on the properties of concrete (Reference 3). Considering the data cited in I&M's submittals (References 1, 3) and provided in an ORNL report (Reference 4), the NRC staff concurs with the licensee's conclusion that the peak SFP temperature above 150 °F, but not exceeding 180 °F, for about 33 days under the worst-case scenario of core offload, will not adversely affect the integrity of the concrete in SFP.

2.3.3 Conclusion

The NRC staff has reviewed the information provided in I&M's letter of May 17, 2001, as supplemented by their letter of September 5, 2001, (References 1, 3) in support of its request to amend Appendix A, Technical Specifications of Facility Operating License DPR-58 and DPR-74, pursuant to 10CFR 50.90. In connection with the SFP structural evaluation for revising the above TS, I&M requested approval to use ACI-349-97 to establish acceptability of the peak SFP temperature with respect to the concrete integrity, since ACI-318-63, cited in the CNP UFSAR does not provide any limitations with respect to maximum short term concrete temperature. On the basis of its review of I&M's submittal, the staff concurs with the licensee's conclusion that the peak SFP temperature above 150 °F, but not exceeding 180 °F, for about 33 days under the worst-case scenario of core offload will not adversely affect the integrity of the concrete in the SFP. Therefore, I&M is justified in its use of ACI-349-97 to establish acceptability of the peak SFP temperature with respect to the concrete integrity, in connection with its request to revise TS 3/4.9.3, "Decay Heat Time."

3.0 STATE CONSULTATION

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

4.0 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 or change the surveillance requirements. 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 44174). 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.

5.0 CONCLUSION

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

6.0 REFERENCES

1. Letter dated May 17, 2001, from M.W. Rencheck, I&M Power, to U.S. NRC. Subject: Donald C. Cook Nuclear Plant Units 1 and 2 - TECHNICAL SPECIFICATION CHANGE REQUEST - REFUELING OPERATIONS DECAY TIME.

2. USNRC Standard Review Plan, NUREG-0800, Section 9.2.5, "Ultimate Heat Sink," including Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling."

3. Letter dated September 5, 2001, from M.W. Rencheck, I&M Power, to U.S. NRC. Subject: Donald C. Cook Nuclear Plant Units 1 and 2 - Response to NRC Request for Additional Information Regarding the Decay Time License Amendment Request .

4. Letter Report No. ORNL / NRR/ LTR / -94/22 titled "Summary of Materials contained in the Structural Materials Information Center," ORNL 10/1994.

5. Letter dated November 29, 2001, from M.W. Rencheck, I&M Power, to U.S. NRC. Subject: Donald C. Cook Nuclear Plant Units 1 and 2 - ADDITIONAL INFORMATION REGARDING TECHNICAL SPECIFICATION REQUEST FOR REFUELING OPERATIONS DECAY TIME.

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