**Indiana Michigan**
**Power Company**
500 Circle Drive
Buchanan, MI 49107 1395

AEP
**INDIANA MICHIGAN POWER**

February 22, 2002
AEP:NRC:2039

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1 and Unit 2
Docket Nos. 50-315 and 50-316
License Amendment Request for Technical Specification 3/4.9, Refueling Operations

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to relocate TS 3/4.9.6, "Refueling Operations – Manipulator Crane Operability" and TS 3/4.9.7 "Refueling Operations - Crane Travel - Spent Fuel Storage Pool Building," with associated Bases to the CNP Updated Final Safety Analysis Report (UFSAR). Relocation of these TS and associated Bases to CNP's UFSAR will provide additional operational flexibility during refueling outages.

The proposed changes are based on the criteria of 10 CFR 50.36(c)(2)(ii) for items requiring a TS limiting condition for operation. The proposed changes are consistent with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 2, dated June 2001.

I&M also proposes format changes to the affected TS page that improve appearance but do not affect any requirements.

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Enclosure 1 provides an oath and affirmation affidavit. Enclosure 2 provides a detailed description and safety analysis to support the proposed changes, including the 10 CFR 50.92(c) evaluation, which concludes that no significant hazard is involved, and the environmental assessment. Attachments 1A and 1B provide marked up TS pages for Unit 1 and Unit 2, respectively. Attachments 2A and 2B provide the proposed TS pages with the changes incorporated for Unit 1 and Unit 2, respectively. Attachment 3 provides a summary of the regulatory commitments made in this submittal.

I&M requests approval of the proposed amendment by April 5, 2002, to support the Unit 1 refueling outage. Once approved, the amendment will be implemented within 30 days.

No previous submittals affect the TS pages that are submitted in this request. If any future submittals affect these TS pages, I&M will coordinate the changes to the pages with the NRC Project Manager to ensure proper TS page control when the associated license amendment requests are approved.

Should you have any questions, please contact Mr. Gordon P. Arent, Manager of Regulatory Affairs, at (616) 697-5553.

Sincerely,

A. C. Bakken, III
Senior Vice President, Nuclear Operations

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Enclosures:
1. Notarized oath and affirmation
2. Evaluation of the proposed changes

Attachments:
1. Unit 1 and Unit 2 Marked-up Technical Specification pages
2. Unit 1 and Unit 2 Proposed Technical Specification pages
3. Summary of commitments made in this letter

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

**AFFIDAVIT**

I, A. Christopher Bakken, III, being duly sworn, state that I am Senior Vice President, Nuclear Operations of American Electric Power Service Corporation and 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.

American Electric Power Service Corporation

A. C. Bakken, III
Senior Vice President, Nuclear Operations

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 22 DAY OF FEBRUARY, 2002

Jennifer L Kernosky
Notary Public

My Commission Expires 5/24/05

**Application for Amendment to Technical Specification (TS) 3/4.9, "Refueling Operations"**

## 1.0 Description

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to relocate TS 3/4.9.6, "Refueling Operations – Manipulator Crane Operability" and TS 3/4.9.7 "Refueling Operations - Crane Travel - Spent Fuel Storage Pool Building," with associated Bases to the CNP Updated Final Safety Analysis Report (UFSAR). Relocation of these TS and associated Bases to CNP's UFSAR will provide additional operational flexibility during refueling outages.

The proposed changes are based on the criteria of 10 CFR 50.36(c)(2)(ii) for items requiring a TS limiting condition for operation (LCO). The proposed changes are consistent with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 2, dated June 2001.

I&M also proposes format changes to the affected TS page that improve appearance but do not affect any requirements.

## 2.0 Proposed Change

I&M proposes that the following Unit 1 and Unit 2 TS and associated Bases be relocated from the CNP TS to the UFSAR:

TS 4/3.9.6, "Manipulator Crane Operability"
TS 4/3.9.7, "Crane Travel – Spent Fuel Storage Pool Building"

Upon relocation to the UFSAR, these requirements would be controlled by 10 CFR 50.59. Additionally, references to these TS will be deleted from the TS Index.

I&M also proposes three types of format changes to the revised TS pages. The changes are:

(1) Reformatting of the headers to include numbered first and second-tier TS section titles and a full-width single line to separate the header section titles from the page text.

(2) Reformatting of the footers to include "Page (page number)" center page, "AMENDMENT (past amendment numbers, with strikethrough, and ending with the current amendment number)" on the right side of the page, and a full-width single line to separate the footer from the page text.

(3) Fully justifying the text and changing the font.

## 3.0 Background

In February 1987, the NRC published an Interim Policy Statement on TS improvements for nuclear power reactors (Reference 1). This policy statement established a specific set of criteria for determining which regulatory requirements and operating restrictions should be included in TS. In November 1987, the Westinghouse Owners Group (WOG) published WCAP-11618 (Reference 2), in which the criteria contained in the Interim Policy Statement were applied to standard Westinghouse TS to determine whether individual specification should be removed and relocated to licensee controlled documents. As documented in WCAP-11618, the WOG determined that the requirements of TS 3/4.9.6 and TS 3/4.9.7 were among those that could be relocated to another controlled document. In a May 1988 letter (Reference 3), the NRC published its conclusions regarding the WOG determinations documented in WCAP-11618. That letter documented NRC agreement with the WOG conclusions regarding relocation of TS 3/4.9.6 and TS 3/4.9.7.

In July 1993, the NRC published its Final Policy Statement (Reference 4) on TS improvements. This policy statement provided four specific criteria for determining which design features and information should be located in the TS as LCOs. These four criteria were very similar to the criteria published in the Interim Policy Statement. The four criteria provided by the Final Policy Statement are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The Final Policy Statement noted that those LCO's that do not meet any of the four criteria may be proposed for removal from the TS and relocated to a licensee-controlled document, such as the UFSAR. The Policy Statement also noted that licensees submitting amendment requests should identify the location and controls for the relocated requirements. The four criteria provided in the Final Policy Statement were codified in 10 CFR 50.36(c)(2)(ii).

## 4.0 Technical Analysis

Provided below is an evaluation of the requirements of TS 3/4.9.6 and TS 3/4.9.7 against the four criteria defined in the NRC's Final Policy Statement and 10 CFR 50.36(c)(2)(ii).

TS 3/4.9.6 Refueling Operations – Manipulator Crane Operability

TS 3/4.9.6 specifies the minimum capacity and overload cutoff limits for the manipulator crane used for the movement of fuel assemblies within the reactor pressure vessel, and specifies the minimum capacity and load indicator limits for the auxiliary hoist used for the movement of control rods within the reactor pressure vessel.

TS 3/4.9.6 ensures that: (1) the manipulator crane will be used for movement of control rods and fuel assemblies within the reactor pressure vessel; (2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and (3) the core internals and pressure vessel are protected from excessive lifting force in the event that they are inadvertently engaged during lifting operations.

Evaluation against 10 CFR 50.36(c)(2)(ii) criteria:

1. The refueling equipment and associated instrumentation is not used to detect, or indicate in the control room, a significant degradation of the reactor coolant pressure boundary.

2. The refueling equipment and associated instrumentation are not process variables, design features, or operating restrictions that are initial conditions of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The applicable design basis accident is a fuel handling accident in which a fuel assembly is dropped, resulting in a release of radioactive material. The refueling equipment and associated instrumentation do not affect the assumptions or initial conditions in the analysis of that accident.

3. The refueling equipment and associated instrumentation are not part of the primary success path to mitigate the consequences of a design basis accident. These components serve no mitigative function.

4. As summarized in Appendix A to WCAP-11618, the requirements of TS 3/4.9.6 were determined not to be risk dominant based on core melt and health effects screening criteria. Additionally, TS 3/4.9.6 does not govern a system, structure, or component requiring risk review/unavailability monitoring as described in CNP's Maintenance Rule Program. The components associated with TS 3/4.9.6 were not evaluated as risk contributors in the CNP Individual Plant Examination (IPE).

Based on the above, the design features and information in TS 3/4.9.6 does not meet the criteria in 10 CFR 50.36(c)(2)(ii) for inclusion as a TS LCO and therefore may be relocated to the CNP UFSAR.

TS 3/4.9.7 Refueling Operations - Crane Travel – Spent Fuel Storage Pool Building

TS 3/4.9.7 prohibits loads in excess of 2,500 pounds from traveling over fuel assemblies in the storage pool. The TS also limits loads carried over the spent fuel pool and the heights at which they may be carried over the racks containing irradiated fuel assemblies so as to prelude impact energies over 24,240 inch-pounds if the loads are dropped from the crane.

TS 3/4.9.7 ensures that, in the event of a dropped load, 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array.

Evaluation against 10 CFR 50.36(c)(2)(ii) criteria:

1. Crane travel limits are not used to detect, or indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. This TS applies to the crane and its interlocks which have both design features and operating restrictions in place to prevent dropping a load on racks containing irradiated fuel that is stored in the spent fuel pool. Criterion 2 requires the design features or operating restrictions to be initial conditions of the design basis accident. The initial condition of the design basis fuel handling accident is the dropping of a single fuel assembly. The crane interlocks are design features that are in place to prevent exceeding the initial condition (i.e. damage to more than one fuel assembly), not an initial condition of itself. These design features are not, in themselves, initial conditions of a design basis accident. Similarly, the load and impact energy limits are operational restrictions that are intended to prevent exceeding the initial condition of the design basis accident. Therefore, the crane, its interlocks, and the load and impact energy limits are provided to prevent operation in a condition that could lead to an unanalyzed load drop accident.

3. The load and impact energy limits and the crane travel interlocks are not part of the primary success path to mitigate the consequences of a design basis accident or transient. These components serve no mitigative function.

4. As summarized in Appendix A to WCAP-11618, the requirements of TS 3/4.9.7 were determined not to be risk dominant based on core melt and health effects screening criteria. Additionally, TS 3/4.9.7 does not govern a system, structure, or component requiring risk review/unavailability monitoring as described in CNP's Maintenance

Rule Program. The components associated with TS 3/4.9.7 were not evaluated as risk contributors in the CNP IPE.

Based on the above, the design features and information in TS 3/4.9.7 does not meet the criteria in 10 CFR 50.36(c)(2)(ii) for inclusion as a TS LCO and therefore may be relocated to the CNP UFSAR.

The relocation of TS 3/4.9.6 and TS 3/4.9.7 is also consistent with NUREG-1431, "Standard Technical Specifications - Westinghouse plants" (Reference 5). NUREG 1431 does not include any specifications equivalent to TS 3/4.9.6 and TS 3/4.9.7. Therefore, relocation of the requirements of TS 3/4.9.6 and TS 3/4.9.7 is consistent with both past and current regulatory positions and requirements.

## 5.0 Regulatory Safety Analysis

### 5.1 No Significant Hazards Consideration

I&M has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No

The proposed changes are administrative in nature in that they result in relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases, to the CNP UFSAR. Changes to the UFSAR are controlled by 10 CFR 50.59. Regulation 10 CFR 50.59 requires that NRC approval be obtained prior to any change to the UFSAR that would result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from changes involving unapproved increases in the probability of occurrence of an accident. The relocation of the requirements of TS 3/4.9.6 and 3/4.9.7 would not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of CNP or the manner in which it is operated. Therefore, the proposed change does not significantly increase the probability of occurrence of an accident previously evaluated.

The proposed change to relocate TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR does not impact the consequences of an accident because there is no effect on the structures, systems and components that mitigate the effects of an accident, or the manner in which they are operated. In accordance with 10 CFR 50.59, if any proposed change to the UFSAR results in more than a minimal increase in the consequences of an accident previously evaluated, NRC review and approval is required prior to the change being made. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from changes involving unapproved increases in the probability of in the consequences of an accident. Therefore, the relocation of requirements will not affect offsite doses, and the consequences of an accident previously evaluated are not significantly increased.

The format changes improve the appearance of the affected pages but do not affect any requirements.

Therefore, the probability of occurrence and the consequences of an accident previously evaluated are not significantly increased.

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

Response: No

The proposed change to relocate TS 3/4.9.6 and 3/4.9.7, with associated Bases, to the CNP UFSAR does not create new accident causal mechanisms. Plant operation will not be affected by the proposed change and no new failure modes will be created. Regulation 10 CFR 50.59 requires that NRC approval be obtained prior to any change to the UFSAR that would create the possibility of a new or different kind of accident from any accident previously evaluated. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from unapproved changes involving new or different kinds of accidents.

The format changes improve the appearance of the affected pages but do not affect any requirements.

Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

Response: No

The proposed change to relocate the requirements from the TS to the UFSAR does not impact equipment design or operation and no changes are being made to the TS required safety limits, safety system settings, or any safety margins associated with TS 3/4.9.6 and 3/4.9.7. Changes to the UFSAR are controlled under the 10 CFR 50.59 process, which requires a safety evaluation to be performed. If any proposed change to the UFSAR results in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered or results in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, NRC review and approval will be required prior to the change being made. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from changes involving a reduction in the margin of safety. The format changes improve the appearance of the affected pages but do not affect any requirements.

Therefore, there is no significant reduction in the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed change involves no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

5.2.1 TS and Regulations

TS 3/4.9.6, TS 3/4.9.7 and their associated Bases are affected by the proposed change in that they would be deleted and their requirements relocated to the UFSAR. Relocation of these requirements is acceptable for the reasons described above.

10 CFR 50.36(c)(2)(ii) specifies the criteria for inclusion of requirements in Technical Specification LCOs. As described above, relocation of the TS 3/4.9.6, and TS 3/4.9.7 requirements to the UFSAR is consistent with 10 CFR 50.36(c)(2)(ii).

5.2.2 Design Bases UFSAR 14.2.1, Fuel Handling Accident

The UFSAR states that the possibility of a fuel handling accident is very remote because of the many administrative controls and physical limitations imposed on the fuel handling operations. Interlocks prevent movement of the crane hook over the spent fuel pool, except when it is necessary to service the pool and its equipment and instrumentation, and to add or remove any equipment associated with spent fuel handling, storage, or inspection. The crane hook is limited to the TS 3.9.7 value with the entire operation under strict administrative control.

The proposed changes do not alter these requirements.

In conclusion, based on the considerations discussed above, (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 Environmental Considerations

I&M has evaluated this license amendment request 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 the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 7.0 Precedent Licensing Actions

In a letter dated January 17, 2002, the NRC issued Amendment No. 137 (Reference 6) to Facility Operating License No. NPF-457 for the Hope Creek Generating Station. This amendment included relocation of TS 3/4.9.6, "Refueling Operations, Refueling Platform," TS 3/4.9.7, "Refueling Operations, Crane Travel - Spent Fuel Storage Pool," and the associated Bases, to the plant's UFSAR.

In a letter dated February 10, 2000, the NRC issued Amendment No. 240 (Reference 7) to Facility Operating Licenses No. DPR-65 for the Millstone Nuclear Power Station, Unit 2. This amendment included relocation of TS 3/4.9.6, "Refueling Operations, Crane Operability-Containment Building," TS 3/4.9.7, "Refueling Operations, Crane Travel - Spent Fuel Storage Building," and the associated Bases, to the plant's Technical Requirements Manual.

Similar to Hope Creek and Millstone Unit 2, I&M proposes to relocate Unit 1 and Unit 2 TS 3/4.9.6 and 3/4.9.7 and their associated Bases to a licensee controlled document. Based on issuance of Hope Creek Amendment No. 137 and Millstone 2 Amendment No. 240, the NRC has determined that the requested change is acceptable.

## 8.0 References

1. 52 FR 3788, "Nuclear Regulatory Commission - Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," dated February 6, 1987.
2. WCAP–11618, "Methodically Engineered, Restructured and Improved, Technical Specifications," dated November 1987.
3. Letter from T. E. Murley, NRC, to R. A. Newton, WOG, dated May, 9, 1988
4. "Nuclear Regulatory Commission - Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993.
5. NUREG 1431, "Standard Technical Specifications Westinghouse Plants," Revision 2, dated June, 2001
6. PSE&G Nuclear, Hope Creek Generating Station, License Amendment No. 137, dated January 17, 2002.
7. Northeast Nuclear Energy Company, Millstone Nuclear Power Plant, Unit 2, License Amendment No. 240, dated February 10, 2000.

Attachment 1A to AEP:NRC:2039

TECHNICAL SPECIFICATION PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 1

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XIII
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3/4 9-8
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~~MANIPULATOR CRANE OPERABILITY~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.6 ~~The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:~~ DELETED

~~a. The manipulator crane used for movement of fuel assemblies having:~~

~~1. A minimum capacity of 3250 pounds, and~~

~~2. An overload cut off limit ≤ 2850 pounds.~~

~~b. The auxiliary hoist used for movement of control rods having:~~

~~1. A minimum capacity of 700 pounds, and~~

~~2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.~~

~~APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.~~

~~ACTION:~~

~~With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.6.1 Each manipulator crane used for movement of fuel assemblied within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.~~

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~~CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.7 ~~Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in. lbs., if the loads are dropped from the crane.~~ DELETED

~~APPLICABILITY: With fuel assemblies in the storage pool.~~

~~ACTION:~~

~~With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.7.1 Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.~~

~~This Surveillance Requirement is not required during the movement of steam generator sections in the auxiliary building for the Unit 1 steam generator replacement project. When crane travel interlocks are disengaged, administrative controls shall be in place to prevent loads from passing over the spent fuel pool.~~

~~4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be ≤ 24,240 in. lbs. prior to moving each load over racks containing fuel.~~

~~* Shared system with Cook Nuclear Plant - Unit 2~~

3/4.9.6 ~~MANIPULATOR CRANE OPERABILITY~~ DELETED

~~The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies 2) each crane has sufficient load capacity to lift a control rod or fuel assembly and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

3/4.9.7 ~~CRANE TRAVEL - SPENT FUEL STORAGE BUILDING~~ DELETED

~~The restriction on movement of loads in excess of 2,500 lbs. over other fuel assemblies in the storage pool ensures that, in the event of a dropped load, 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. The 2,500 lb. load restriction is based on the combined nominal weight of a fuel assembly, a control rod assembly, and an associated fuel handling tool. Release of activity from a single fuel assembly is consistent with the assumption for the analysis for a fuel handling accident.~~

~~The restriction on movements of loads in excess of the impact energy limit, which is based on the kinetic energy of a dropped fuel assembly and control rod assembly from 15" above the fuel storage rack, is to bound other loads.~~

~~Prohibiting loads greater than 2,500 pounds or loads at heights that would exceed the kinetic energy impact limit allows flexibility in the movements of fuel and other relatively light loads, while providing reasonable assurance that the consequences of a fuel handling accident will not be exceeded.~~

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

Attachment 1B to AEP:NRC:2039

TECHNICAL SPECIFICATION PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 2

X
XIII
3/4 9-6
3/4 9-7
B3/4 9-2

~~MANIPULATOR CRANE OPERABILITY~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.6 ~~The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:~~ DELETED

~~a. The manipulator crane used for movement of fuel assemblies having:~~

~~1. A minimum capacity of 3250 pounds, and~~

~~2. An overload cut off limit ≤ 2850 pounds.~~

~~b. The auxiliary hoist used for movement of control rods having:~~

~~1. A minimum capacity of 700 pounds, and~~

~~2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.~~

~~APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.~~

~~ACTION:~~

~~With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.~~

~~4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.~~

~~CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.7 ~~Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in. lbs., if the loads are dropped from the crane.~~ DELETED

~~APPLICABILITY: With fuel assemblies in the storage pool.~~

~~ACTION:~~

~~With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.7.1 Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.~~

~~This Surveillance Requirement is not required during the movement of steam generator sections in the auxiliary building for the Unit 1 steam generator replacement project. When crane travel interlocks are disengaged, administrative controls shall be in place to prevent loads from passing over the spent fuel pool.~~

~~4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be ≤ 24,240 in. lbs. prior to moving each load over racks containing fuel.~~

~~* Shared system with Cook Nuclear Plant - Unit 1.~~

3/4.9.6 ~~MANIPULATOR CRANE OPERABILITY~~ DELETED

~~The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

3/4.9.7 ~~CRANE TRAVEL - SPENT FUEL STORAGE BUILDING~~ DELETED

~~The restriction on movement of loads in excess of 2,500 lbs. over other fuel assemblies in the storage pool ensures that, in the event of a dropped load, 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. The 2,500 lb. load restriction is based on the combined nominal weight of a fuel assembly, a control rod assembly, and an associated fuel handling tool. Release of activity from a single fuel assembly is consistent with the assumption for the analysis for a fuel handling accident.~~

~~The restriction on movements of loads in excess of the impact energy limit, which is based on the kinetic energy of a dropped fuel assembly and control rod assembly from 15" above the fuel storage rack, is to bound other loads.~~

~~Prohibiting loads greater than 2,500 pounds or loads at heights that would exceed the kinetic energy impact limit allows flexibility in the movements of fuel and other relatively light loads, while providing reasonable assurance that the consequences of a fuel handling accident will not be exceeded.~~

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

Attachment 2A to AEP:NRC:2039

# PROPOSED TECHNICAL SPECIFICATION PAGES

REVISED PAGES
UNIT 1

X
XIII
3/4 9-6
3/4 9-7
3/4 9-8
B3/4 9-2

SECTION PAGE

3.9.6 DELETED

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3.9.7 DELETED

3/4.9.6 DELETED

3/4.9.7 DELETED

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

Attachment 2B to AEP:NRC:2039

PROPOSED TECHNICAL SPECIFICATION PAGES

REVISED PAGES
UNIT 2

X
XIII
3/4 9-6
3/4 9-7
B3/4 9-2

3.9.6 DELETED

3.9.7 DELETED

3/4.9.6 DELETED

3/4.9.7 DELETED

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

Attachment 3 to AEP:NRC:2039

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this 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 |
|---|---|
| I&M will revise the appropriate sections of the UFSAR to include requirements and information from TS 3/4.9.6, 3/4.9.7 and associated Bases. | Within 30 days from date of approval of license amendment request. |