UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated September 23, 1999, as supplemented October 11 and November 10, 1999, the Indiana Michigan Power Company (IM, or the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The proposed amendments would provide approval to move steam generator sections through the auxiliary building and to disengage crane travel interlocks, and provide relief from performance of Technical Specification Surveillance Requirement (TSSR) 4.9.7.1. The loads to be moved are in support of the Unit 1 Steam Generator Replacement Project (SGRP). Since the Unit 1 steam generator sections are heavier than those evaluated in the Updated Final Safety Analysis Report (UFSAR) for the auxiliary building crane over the planned load path, the licensee concluded that the proposed activity may increase the probability of occurrence or the consequences of an accident and requested prior NRC approval in accordance with 10 CFR 50.59.

The October 11, 1999, submittal provided corrected TS pages. The November 10, 1999, submittal was in response to a NRC request for additional information dated October 26, 1999, and provided clarifying information to the original submittal. This information was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

2.0 EVALUATION

The Unit 1 original Westinghouse Model 51 steam generators will be replaced with Babcock and Wilcox (B&W) Model 51R steam generators due to the degrading condition of the original steam generator tubes. The steam generator (SG) replacement will involve partial disassembly of the reinforced concrete enclosures surrounding each SG and implementation of a two-piece replacement methodology. This approach includes cutting the SGs into an upper section (steam dome) and lower section (SG lower assembly). Both sections will be removed from containment. The steam domes will be refurbished and returned along with the replacement lower sections supplied by B&W.

The SG sections will be moved between the containment equipment hatch and crane bay using the auxiliary building cranes. On March 8, 1988, the NRC approved TS Amendment No. 100 to Facility Operating License No. DPR-74 for the CNP Unit 2 SGRP. In order to facilitate the Unit 2 SGRP, the licensee modified its existing auxiliary building crane to meet the single-failure-proof criteria of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The licensee performed approved modifications to the auxiliary building to support the modifications to the auxiliary building crane. The licensee also procured an additional single-failure-proof crane for the auxiliary building to be used in tandem with the existing crane in order to support the steam generator section moves through the auxiliary building. This approach was acceptable to the NRC. The load size and weight, handling equipment and methods, and load paths for movement of the Unit 1 SG sections through the auxiliary building are similar to those approved by the NRC for the Unit 2 SGRP in Amendment No. 100. The Unit 2 SGRP moved loads up to approximately 277 tons using the tandem crane arrangement. However, since that approval was only applicable to the Unit 2 SGRP, the licensee made this request for Unit 1. The licensee proposes to move loads up to approximately 270 tons using the tandem crane arrangement for the Unit 1 SGRP. Because the Unit 1 SG sections are heavier than those evaluated for a seismic event in the UFSAR and the proposed activity may increase the probability of occurrence or the consequences of an accident, NRC approval of the proposed load handling is required in accordance with 10 CFR 50.59.

The licensee proposes to (1) perform load handling for 16 SG sections that are heavier than the loads previously evaluated for the proposed load path for CNP's heavy loads program and (2) disengage the crane travel interlocks of TSSR 4.9.7.1 to accommodate movement of the cranes at the southwest corner of the spent fuel pool (SFP). The TS requirements are the same for Unit 1 and Unit 2, and the cranes and SFP are common to both units.

2.1 HANDLING OF HEAVY LOADS CONSISTENT WITH NUREG-0612

In Generic Letter 85-11, "Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612," the NRC concluded that satisfying the NUREG-0612 Phase I guidelines assures that the potential for a load drop is extremely small. The handling of heavy loads at CNP is consistent with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Plants," as concluded by the NRC and documented in a safety evaluation report dated September 20, 1983. The licensee evaluated the proposed handling of the Unit 1 SG sections with respect to the seven guidelines of NUREG-0612 as provided below.

2.1.1 Safe Load Path

Primary reliance for safe load handling during the proposed activity is placed on the use of single-failure-proof cranes; however, the licensee performed a review and walkdown of the load path though the auxiliary building to identify potential interactions with equipment important to safety.

The load drop evaluation included conservative assumptions. In the unlikely event of a drop, the SG section is assumed to penetrate all intervening structures, systems, and components and stop at the building foundation. All components beneath the entire load path are assumed to lose functional capability, regardless of where along the path the drop occurs. For example, when evaluating affected equipment or components in the west end of the load path, no credit is taken for equipment in the east end of the load path to mitigate the event. This is a conservative assumption since a dropped steam generator component is not large enough to impact all equipment below the load path simultaneously.

Postulated load drops were not considered directly over the SFP, because the SG sections do not travel directly over the SFP and the center of gravity of the SG sections is maintained outside of the exclusion area at all times. The load path for Unit 1 SG sections differs from the load path used for the Unit 2 SGRP in 1988. Because of the Unit 1 containment equipment hatch and radiation shield wall, movement of each Unit 2 SG lower assembly required one end of the section to be moved over the southwest corner of the SFP. This is not true for movement of the Unit 1 SG sections. The licensee's evaluation concluded that the integrity of the SFP would be maintained. The conclusion was based on the geometry of the SFP and adjacent building structure elements, as well as the relative member sizes of the SFP wall and connecting building structure elements. Although cracking and localized damage to the 5-feet 2-inch thick reinforced concrete pool wall would occur, the steel liner would remain intact due to its ductility.

If a load drop of a SG section is postulated that results in damage to the SFP cooling piping external to the SFP, a potential loss of SFP cooling could occur. Assuming the maximum design basis heat load in the SFP without an initial loss of water inventory, the design basis analysis for compete loss of SFP cooling demonstrates that bulk boiling would not occur for 5.74 hours. The decay heat of fuel assemblies currently in the SFP is relatively low compared to the maximum values assumed in the design basis analysis. The licensee has calculated that the current time for bulk boiling in the SFP is greater than 31.6 hours from a starting point of 116°F and greater than 22.1 hours from a starting point of 144°F. Therefore, significant time would exist for operator actions to either restore normal SFP cooling or provide alternate cooling methods for a complete loss of SFP cooling. The SFP is designed and maintained to prevent inadvertent draining of the pool if the external SFP cooling piping were damaged. Therefore, adequate SFP water inventory would be ensured even in the event of damage to the external SFP cooling water piping. Mitigation of a loss of SFP cooling event is governed by approved plant operating procedures, which include listing acceptable sources of makeup water to the SFP.

Since the SG sections do not travel directly over the SFP and the center of gravity of the SG sections is maintained outside of the exclusion area at all times, the licensee concluded that potential damage to new or spent fuel assemblies in the SFP cannot result from a direct drop of a section. The licensee also considered the possibility for a dropped component to roll into the SFP. The qualitative evaluation divided the load path into five representative positions for the steam generator sections and considered three orientations for dropping of each section, each end falling first and a horizontal drop. The postulated load drops would result in significant damage to the runway beams or concrete floor in the load path and possible damage to the SG section itself; however, the damage and appurtenances on the SG sections would resist rolling of the loads toward the SFP. The licensee concluded that there was reasonable assurance that movement of the SG sections could be performed without the loads traveling into the SFP.

Although equipment important to safety could be affected, the licensee concluded that the operating unit's safe shutdown and reactor decay heat removal requirements continue to be satisfied. Potential damage to Unit 1 safety-related systems and portions of the common safety-related systems affecting Unit 1 would have no safety significance for the Unit 1 reactor and supporting systems, since removal and replacement of SG components in Unit 1 can only be performed while the Unit 1 reactor is defueled. The licensee evaluation concluded that, with the exception of component cooling water (CCW), potentially affected Unit 2 safety-related systems and portions of common safety-related systems affecting Unit 2 are not needed in order to safely shut down the Unit 2 reactor in the event of a load drop. There is no possibility of a load drop directly affecting reactor coolant system pressure boundary piping or creating an accident for the operating Unit 2 reactor. The most critical Unit 2 system that could be affected by a load drop in the auxiliary building is CCW. Unit 2 CCW equipment potentially damaged by a load drop includes the supply and return lines to the associated SFP heat exchangers. These lines are part of the miscellaneous CCW header supplied from one of the separate trains of CCW. Damage to this piping could result in temporary loss of both trains of Unit 2 CCW until the ruptured pipe on the miscellaneous CCW header could be isolated. The isolation valves for this piping are located outside the load path. Once the ruptured piping on the miscellaneous CCW header is isolated, complete support of residual heat removal would be restored. Existing plant procedures already provide for isolation of the affected sections of CCW piping. Therefore, this potential failure would not prevent safe shutdown and adequate decay heat removal for Unit 2.

The licensee evaluated the potential release of radioactive materials from sources other than the SFP. The evaluation determined that the only potential release of liquid radioactive material from load drop is a rupture of the radioactive waste holding tanks in the auxiliary building. As documented in UFSAR chapter 14.2.2, any spillage of fluid due to a tank rupture would drain to the sump tank or waste holdup tanks or would accumulate in the sump areas. Prior to release to the environment, sampling is required to ensure that discharge is within licensed limits. The postulated load drop could rupture waste gas vent lines to the suction header of the waste gas compressors. This would result in the release of a small amount of radioactive gas, but would not result in the release of any of the contents of the waste gas decay tanks. A rupture of one waste gas decay tank has been evaluated in UFSAR chapter 14.2.3. Therefore, the licensee concluded that the consequences of the current design basis waste decay tank rupture bound the postulated load drop.

As stated in their November 10, 1999, submittal, the licensee intends to implement the following measures prior to performing the SG load handling in the auxiliary building in order to mitigate the consequences of a potential load drop.

- An operator briefing for response to a load drop will be completed prior to movement of SG sections in the auxiliary building. This briefing will highlight the equipment potentially impacted by dropping of a SG section and the applicable response as defined in existing procedures.
- No movement of fuel assemblies will be allowed in either Unit 1 or Unit 2 containment buildings, or in the auxiliary building.
- The SFP will be isolated from Unit 2 containment.
- The weir gate between the SFP and the fuel transfer canal will be closed and pressurized.
- The SFP area exhaust fans will be required to be operable.

- Any additional items that result from compliance with the approved plant procedure for conducting infrequently performed evolutions will be incorporated into the specific heavy load procedure that governs the movement of SG sections in the auxiliary building.

Based on the above evaluation, the staff concludes that safe load paths have been implemented in a manner consistent with NUREG-0612 and are acceptable.

2.1.2 Load Handling Procedures

The licensee intends to provide load handling procedures that are specific to the upper and lower SG sections. Load paths will be defined within the load handling areas and qualified personnel will direct the crane operator to ensure conformance to the prescribed load path. The procedures will also address equipment identification, inspection and acceptance criteria, step-by-step load handling sequences, and special precautions.

The staff concludes that load handling procedures will be implemented in a manner consistent with NUREG-0612 and are acceptable.

2.1.3 Operator Training

The licensee stated that crane operators are trained and qualified using maintenance skills training lesson plans that include the requirements of American National Standards Institute (ANSI) B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)", Chapter 2-3, "Qualifications for Operators." Station-qualified crane operators used for the SG section lifts will receive both classroom and hands-on training based on these lesson plans. Training will include orientation with the specific procedures to be used for the SG section lifts prior to beginning the corresponding crane operations.

The staff concludes that crane operator training will be implemented in a manner consistent with NUREG-0612 and is acceptable.

2.1.4 Special Lifting Devices

The licensee stated that its heavy loads program includes the use of special lifting devices and requires design, fabrication, and testing that provide load handling reliability consistent with ANSI N14.6-1978, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." The special lifting devices used for the Unit 2 SGRP will be used for Unit 1. Prior to use, the devices will be inspected to verify there has been no significant corrosion or structural distress and tested in accordance with ANSI N14.6-1978.

The staff concludes that design and testing of special lifting devices will be implemented in a manner consistent with NUREG-0612 and is acceptable.

2.1.5 Lifting Devices (Not Specially Designed)

The licensee stated that standard lifting devices used for the movement of the SG sections will be selected and used in accordance with the guidelines of ANSI B30.9-1996, "Slings." Due to the relatively slow hoist speeds of the cranes at CNP subject to NUREG-0612, the NRC previously concluded in its September 20, 1983, safety evaluation report that the dynamic loads imposed on these slings are reasonably small and may be disregarded when determining the static load to be used when selecting and using slings.

The staff concludes that installation and use of slings is consistent with NUREG-0612 and is acceptable.

2.1.6 Cranes (Inspection, Testing, and Maintenance)

The licensee stated that the auxiliary building cranes at CNP are inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976. In preparation for the Unit 1 SGRP, the cranes will be inspected to confirm consistency with the single-failure-proof guidelines of NUREG-0612 and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

The staff concludes that crane component inspection and testing is consistent with NUREG-0612 and is acceptable.

2.1.7 Crane Design

The licensee stated that both auxiliary building cranes were determined to be acceptable for ANSI B30.2-1976 and Crane Manufacturers Association of America (CMAA)-70-1975, "Specifications for Electric and Overhead Traveling Cranes."

The staff concludes that crane design is consistent with NUREG-0612 and is acceptable.

2.2 CONSISTENCY WITH NUREG-0554

The licensee confirmed that the auxiliary building cranes are designed to meet the single-failure-proof criteria of NUREG-0554. The west auxiliary building crane has a design rated load (DRL) capability of 150 tons and a maximum critical load (MCL) capability of 55 tons. The east auxiliary building crane has a DRL of 150 tons and MCL of 60 tons. The two auxiliary building cranes were modified for tandem configuration as a lift system to handle the eight lifts of the Unit 2 SGRP. The tandem configuration retains the single-failure-proof features of the individual cranes and provides a DRL of 300 tons. The NRC previously found that the licensee has demonstrated that the cranes meet the single-failure-proof criteria of NUREG-0554, as documented in the safety evaluation report for Amendment No. 100 for Unit 2.

2.3 SEISMIC CONSIDERATIONS

The SG sections are considered critical loads per NUREG-0554 because they are brought through the auxiliary building in the vicinity of the SFP. For these loads, the design basis seismic capability of the load handling equipment and structures, as analyzed in the UFSAR, is exceeded. Notwithstanding the low probability of an SSE (safe shutdown earthquake) during the movement of the SG sections, the licensee studied the design adequacy of the auxiliary building cranes and structure for the tandem crane 300-ton design rated load to demonstrate that the load is safely retained even in the event of an SSE.

The engineering study follows design basis methodology except in two aspects. First, the pendulum effect of the lifted load was incorporated into the analysis, thereby determining realistic effects of the seismic accelerations of the load on the crane and supporting structure. Second, seismic vertical response spectra were generated instead of the design basis assumption of 2/3 of the horizontal spectra at the crane rail elevation. A mass and vertical stiffness mathematical model was developed that follows the criteria and methodology described in the UFSAR for the development of the design basis seismic horizontal response spectra. The design basis includes conservatism because it ignores vertical soil-structure interaction and the significantly stiffer, nearly seismically rigid, vertical dynamic response of the auxiliary building. Using calculated seismic vertical response spectra produces a large reduction in the maximum spectral amplitude compared to the design basis. Using the more realistically calculated seismic vertical response spectra, the licensee engineering study determined that the resultant wheel loads associated with the tandem crane, a 300 ton load, and an SSE are bounded by those previously evaluated in the existing CNP design basis. The NRC previously reviewed the two cranes and the auxiliary building structure for the Unit 2 SGRP (Unit 2 Amendment No. 100) and concluded that the resulting stresses are below the allowable values for the conditions imposed by two 150-ton single-failure-proof cranes lifting the steam generator lower assembly (the heaviest component) and by the combined 115-ton load during the SSE. As part of CNP's larger effort to improve design basis documentation, the engineering study is being reviewed. It is not expected that the review will impact the study's conclusions, but the licensee intends to complete the study prior to moving SG sections.

A telephone call was held on November 30, 1999, between F. Lyon and B. Thomas (NRC) and L. Lahti, W. MacRae, S. McBee, and others (IM) to clarify the information contained in the September 23, 1999, application. An additional telephone call was held on December 3, 1999, between F. Lyon and B. Jain (NRC) and J. Burford, W. MacRae, S. McBee, and others (IM) to clarify the information contained in the September 23, 1999, application. In the discussions, the licensee confirmed that design margins in the rope tension, crane bridge stress, supporting concrete structures stresses, and the wheel loads are within the existing design basis for CNP. The licensee stated that it considered variation in lifted loads and assured itself that the variation in crane's natural frequencies due to lifted loads of less than 300 tons will not reduce the design margins in various crane components and supporting structures. The licensee also stated that in the development of the vertical seismic spectra at the crane level, soil-structure-interaction effects were modeled in accordance with the staff's guidelines for SEP [systematic evaluation program] facilities contained in NRC letter LS05-80-12-035 from D. Crutchfield (NRC) to all SEP facilities, dated December 15, 1980. The vertical seismic model was analyzed for two foundation seismic motion input spectra shapes: the Housner's spectra, and the NUREG-0098 spectra. The licensee enveloped the vertical spectra at the crane level from the two seismic inputs. The licensee also stated their intent to verify that movement resulting from the pendulum motion of a seismic event would not impact safety-related equipment.

The runway beam system used for moving the loads through the containment equipment hatch uses a simple design of carts with guided rollers. The structure is temporary and is supported by the containment equipment hatch and the auxiliary building and containment building floor slabs. The licensee evaluated the runway beam system for the static and dynamic loads imposed by the SG sections, cart, and rigging. Similar to the Unit 2 SGRP, the loaded runway beam system was not evaluated for seismic loads, but the design provides a defined travel path that is located within an evaluated auxiliary building load handling area. Consistent with NUREG-0612 safe load path guidelines, this minimizes the potential for impacting equipment important to safety and ensures that the requirements for safe shutdown, decay heat removal, and SFP cooling continue to be met in the event of a load drop.

Based upon the cranes meeting the single-failure-proof criteria of NUREG-0554 and the previous NRC evaluation and approval for the movement of heavy loads during the Unit 2 SGRP, the staff finds the results of the licensee's seismic evaluation reasonable.

2.4 TECHNICAL SPECIFICATIONS

The licensee proposes to disengage the crane travel interlocks to accommodate movement of the cranes at the southwest corner of the SFP and to change TSSR 4.9.7.1 by adding the statement, "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."

TS 3.9.7 prohibits the movement of loads in excess of 2500 pounds over fuel assemblies in the SFP. Associated TSSR 4.9.7.1 requires that the crane interlocks that limit crane travel and help ensure compliance with TS 3.9.7 are demonstrated operable within seven days of crane use and at least once per seven days thereafter.

The TS Bases state that the restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident, consistent with the activity release assumed in the accident analysis. The licensee states that, since no portion of the steam generators will pass over any part of the SFP, TS 3.9.7 will be met during the proposed load handling. The center of gravity of the rigging assembly remains outside of the SFP exclusion zone, which consists of the SFP and an additional margin beyond the border of the pool. Maintaining the center of gravity at this distance provides additional assurance that a drop of the rigging assembly would not result in impact to spent fuel assemblies, thus meeting TS 3.9.7 and the purpose of the associated interlocks. Furthermore, NUREG-0612, Section 5.1.2, states that meeting the single-failure-proof criteria of NUREG-0554 is a satisfactory alternative to crane travel interlocks. Procedural controls discussed in Sections 2.1.1, 2.1.2, and 2.1.3 above provide reasonable assurance to prevent loads from passing over the SFP.

Therefore, the staff finds that the proposed change to TSSR 4.9.7.1 is acceptable.

2.5 CONCLUSION

Based on the above evaluation, the staff finds that the movement of steam generator sections and disengaging the crane travel interlocks of TSSR 4.9.7.1 to accommodate movement of the cranes for the Unit 1 SGRP are acceptable.

3.0 STATE CONSULTATION

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

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 (64 FR 57665). 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 staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Lyon

Date: December 7, 1999