test to be acceptable provided the general containment inspection (10 CFR Part 50, Appendix J, Section V.A.) is performed.

Pursuant to 10 CFR 51.32, the Commission has determined that granting this Exemption will not have a significant impact on the environment (60 FR 39020).

This Exemption is effective upon issuance and shall expire after March 31, 1997, or at the completion of the 1997 refueling outage whichever comes first.

Dated at Rockville, Maryland, this 3rd day of August 1995.

For the Nuclear Regulatory Commission.

## Elinor G. Adensam,

Deputy Director, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

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## [Docket No. 50-315]

## Indiana Michigan Power Co.; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR– 58, issued to Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook Nuclear Plant, Unit 1, located in Berrien County, Michigan.

The proposed amendment would modify technical specifications 4.4.5.4 and 4.4.5.5, on steam generators, to allow for repair of hybrid expansion joint sleeves under redefined repair boundary limits.

The licensee requested this change on an exigent basis because: (1) The change is associated with steam generator tube repairs during the Unit 1 refueling outage currently in progress, and (2) the empirical data compiled from the Kewaunee Nuclear Plant steam generator tube pulls in March 1995 is the primary support for this amendment and the final implications and conclusions from assessment of that data are just now being formulated. The Unit 1 tube repairs are currently scheduled to begin on August 29, 1995.

The NRC staff has reviewed and concurred with the licensee's reasons for requesting this amendment on an exigent basis. Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

Pursuant to 10 CFR 50.91(a)(6), for amendments to be granted under exigent circumstances the NRC staff must determine that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the CNP [Donald C. Cook Nuclear Plant] unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Mechanical testing has shown that the inherent structural strength of the HEJ [hybrid expansion joint] provides sufficient integrity such that the tube rupture capability recommendations of RG [Regulatory Guide] 1.121 are met, even for instances of 100% throughwall, 360° circumferentially oriented degradation in the HEJ hardroll lower transition region. Structural integrity recommendations consistent with RG 1.121 are supplied for all tube degradation 1.1 inch or greater below the bottom of the HEJ hardroll upper transition. Based on test data, a bounding SLB [steam line break] leak rate of 0.033 gpm for indications between 1.1 and 1.3 inch below the bottom of the hardroll upper transition is applied. As the leakage data base is expanded and statistical basis established, this SLB leakage allowance may be reduced. For indications existing greater than 1.3 inch below the bottom of the hardroll upper transition, SLB event leakage can be neglected.

Additional prevention from tube rupture is inherently provided by the HEJ geometry. For RCS [reactor coolant system] release rates to exceed the normal makeup capacity of the plant, approximately 120 gpm, the tube must be postulated to experience a complete circumferential separation at the lower transition, and become axially displaced by 3 to 3.25 inches, resulting in complete geometric disassociation between the tube and sleeve resulting in sufficient flow area to support leakage of 120 gpm. During the 1989 plug top release event at North Anna unit 1, primary to secondary release rates were calculated to be less than 80 gpm, for a flow area approximately 4 times larger than the flow area created by a tube which has axially

displaced by about 1.25 to 1.5 inch. Analysis of the steam generator indicates that at a 95% cumulative probability, the tube would experience an axial displacement of less than the 1.1 inch boundary. At this level of axial displacement, a ring of metal to metal contact would remain between the tube and sleeve, and leakage would be far less than 120 gpm. Projected leakage at this point is expected to be less than 2.5 gpm. Therefore, implementation of the proposed repair boundary will not result in tube rupture, even for a tube postulated to not behave as predicted by the available test and pulled tube data.

The proposed technical specification change to support the implementation of the HEJ sleeve tube repair boundary for parent tube degradation in the HEJ hardroll lower transition region does not adversely impact any other previously evaluated design basis accident or the results of accident analyses for the current technical specification minimum reactor coolant system flow rate. Plugging limit criteria are established using the guidance of RG 1.121. Furthermore, per RG 1.83 recommendations, the sleeved tube assembly can be monitored through periodic inspections with present eddy current techniques.

(2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the repair boundary will not introduce significant or adverse changes to the plant design basis. Mechanical testing of degraded sleeve joints supports the conclusions of the calculations that the sleeve retains structural (tube burst) capability consistent with RG 1.121. As with [the] initial installation of sleeves, implementation of the alternate criteria cannot interact with other portions of the RCS. Any hypothetical accident as a result of potential tube degradation in the HEJ hardroll lower transition region of the tube is bounded by the existing tube rupture accident analysis. Neither the sleeve design nor implementation of the tube repair boundary defined in Attachment 4 Westinghouse Electric Corporation Proprietary Report, WCAP-14446] affects any other component or location of the tube outside of the immediate area repaired. In addition, as the installation of sleeves and the impact on current plugging level analyses is accounted for, any postulation that the alternate repair criteria for parent tube degradation in the HEJ hardroll lower transition creates a new or different type of accident is not supported.

(3) The proposed license amendment does not involve a significant reduction in a margin of safety.

The safety factors used in the establishment of the HEJ sleeved tube alternate repair boundary for the disposition of indications in the hardroll lower transition of potentially degraded parent tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. Based on the sleeved tube geometry, it is unrealistic to consider that application of the repair boundary could result in single tube leak