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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Responds to Generic Ltr 84-04 re elimination of postulated pipe breaks in PWR primary main loops. Elimination of large RCS primary loop pipe breaks from consideration in structural design basis requested to resolve open item.

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September 10, 1984
AEP:NRC:0137D

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NRC GENERIC LETTER 84-04; ELIMINATION OF POSTULATED
PIPE BREAKS IN PRIMARY MAIN LOOPS
GENERIC ISSUE A-2, ASYMMETRIC BLOWDOWN LOADS ON PWR PRIMARY SYSTEMS
REQUEST FOR LICENSE CONDITION DELETION

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

References:
1. WCAP 9558, Revision 2 (May 1981), "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack."
2. WCAP 9787 (May 1981) "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation."
3. Letter Report NS-EPR-2519, E.P. Rahe to D. G. Eisenhut (November 10, 1981), Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981.
4. Letter No. AEP:NRC:0137C dated October 8, 1980.

Dear Mr. Denton:

This letter and its Attachment are in response to the NRC Generic Letter No. 84-04 entitled, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," dated February 1, 1984. The Westinghouse Utility Owners Group, of which Indiana & Michigan Electric Company is a member, submitted the analytical evaluation and mechanistic fracture evaluation of the effect of the Asymmetric Load for the Postulated Primary Loop Pipe Breaks.

USNRC Generic Letter 84-04 provided the staff Safety Evaluation Report for the mechanistic fracture analysis submitted by the Westinghouse Utility Group (Reference Nos. 1 and 2). The staff evaluation concluded that, provided two conditions noted in Generic Letter No. 84-04 were met, an acceptable technical basis exists so that the asymmetric blowdown loads resulting from large breaks in the main coolant loop piping need not be

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considered as a design basis for the sixteen domestic plants for which the analysis applies. The purpose of this letter is to respond to the open items identified in Generic Letter 84-04, to obtain final resolution to Generic Issue A-2 for Units 1 and 2, and to request deletion of the License Condition C.3(a) to Operating License No. DPR-74 for Unit No. 2 of the Donald C. Cook Nuclear Plant.

The Safety Evaluation provided in Generic Letter No. 84-04 requires that two conditions must be met i.e., verification of bending moment loads at two nuclear plants and verification of leak detection capability. The Donald C. Cook Nuclear Plant is not one of the two plants noted in the Generic Letter for which confirmation of the maximum bending moments is required. Thus, the first condition is not applicable to the Donald C. Cook Nuclear Plant. The second condition, that leakage detection systems exist to detect postulated flaws utilizing guidance from Regulatory Guide 1.45, with the exception of seismic equipment qualification for the airborne particulate radiation monitor is applicable.

The Donald C. Cook Nuclear Plant has several leak detection systems as noted in Technical Specification 3.4.6.1. The operability requirements of the leak detection systems under both identified and unidentified leakage conditions are noted in T/S 3.4.6.2. With the leak detection systems noted in T/S 3.4.6.1 in operation, we have the capability of detecting a one gpm unidentified leak within four hours. The leak detection systems at the Cook Plant have been shown to be capable of detecting leaks during feedwater line cracks (Reference No. 3). The leak detection system designs were evaluated against the requirements of item 2 of the Generic Letter. As a result of that review, we believe that the leakage detection systems at Cook Plant are consistent with the requirements of item 2 of the Generic Letter.

The mechanistic fracture mechanics evaluations contained in References 1 and 2 demonstrated that large margins against unstable crack extension exist for the stainless steel primary piping postulated to have large flaws and subjected to postulated safe shutdown earthquake and other plant loadings. The technical evaluation of the NRC staff attached to Generic Letter 84-04 supports the conclusions of the Westinghouse WCAP reports. The staff report further states that the potential for failure is low so that breaks in the main reactor coolant piping up to and including a break equivalent in size to the rupture of the largest pipe, need not be postulated as a design basis for defining structural loads on or within the reactor vessel and the rest of the reactor coolant system main loops. The staff report concludes that these pipe breaks need not be considered as a design basis to resolve generic safety issue A-2, "Asymmetric Blowdown Loads on PWR Primary System". This is applicable to both Units of the Donald C. Cook Nuclear Plant.

License Condition C.3(a) to Operating License No. DPR-74 for Unit No. 2 requires that an analytical evaluation be made of the effects of certain postulated break loads on the reactor coolant system and internals. We have submitted the analytical and mechanistic fracture evaluations in WCAP reports noted in References 1, 2 and 4. The NRC has accepted the results of the above evaluation, and thus, we have satisfactorily completed the

requirements of the license condition. As such, we hereby request that License Condition C.3(a) be deleted from Operating License No. DPR-74 for Unit No. 2 of the Donald C. Cook Nuclear Plant.

The Safety Evaluation Report (SER) from the NRC staff attached to G.L. 84-04 assesses the relative costs and public risk of using advanced fracture mechanics techniques to justify design bases for several operating PWR's instead of modifying these plants to conform to piping restraint designs used in more recent plants. The NRC SER concludes that the dose and cost benefits indicate that not requiring installation of plant modifications to mitigate consequences of asymmetric pressure loads resulting from a possible primary system (Double Ended Guillotine) pipebreak would result in very little increase in public risk and accident avoided occupational exposure (less than 5-man-rem) and would avoid significant plant installation occupational exposure and industry and NRC costs. Based on the above evaluation we believe that this license amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; does not create the possibility of a new accident, or one different from any previously evaluated; and does not involve a reduction in the margin of safety. Therefore the proposed amendment does not involve a significant hazards consideration as defined in 10 CFR 50.92. This license amendment request does not involve an unreviewed safety question and will not significantly effect the amounts of effluents released to the environment. A copy of this request for license condition deletion has been sent to the State of Michigan.

Generic Letter No. 84-04 also states that authorization by the NRC to remove or not to install protection against asymmetric dynamic loads in the primary coolant loop will require an exemption from General Design Criterion No. 4 (GDC-4). Indiana & Michigan Electric Company does not believe that an exemption is required because footnote No. 1 to 10 CFR 50, Appendix A, states, "Further details relating to the type, size and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development." Also, among other things, the introduction paragraph to Appendix A notes that the development of these General Design Criteria is not yet complete. Thus, the existing criteria anticipated that, when developed, justification such as advanced fracture mechanics analyses could be used to define postulated LOCA pipe break sizes less than the double-ended rupture of the largest pipe of the reactor coolant system. The NRC staff concludes in the SER attached to G.L. No. 84-04, that the final criteria will not differ significantly from the evaluation results presented in the above referenced Westinghouse Topical Reports. Nevertheless, since sufficient justification for an exemption has been presented and, for all intents and purposes, an exemption will have no effect on plant operation, Indiana & Michigan Electric Company requests that an exemption from GDC-4 be issued as set forth in the enclosed application.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

9/7/84

M. P. Alexich
Vice President

MPA/cm

Attachments

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
E. R. Swanson, NRC Resident Inspector - Bridgman

ATTACHMENT TO AEP:NRC:0137D

EXEMPTION APPLICATION
INDIANA & MICHIGAN ELECTRIC COMPANY
DONALD C. COOK NUCLEAR PLANT

This exemption application is in response to Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," dated February 1, 1984. Indiana & Michigan Electric Company requests the elimination of large reactor coolant system primary loop pipe breaks from consideration in the structural design basis of the Donald C. Cook Nuclear Power Plant. This request is based upon the use of advanced fracture mechanics technology as applied to primary system piping in Westinghouse Electric Corporation Topical Reports WCAP 9558, Revision 2 (proprietary), and WCAP 9787 (proprietary), and is the resolution of Generic Issue A-2, "Asymmetric Blowdown Loads on PWR Primary Systems."

The bases for the request for exemption are as follows:

1. Extensive operating experience has demonstrated the integrity of the PWR reactor coolant system primary loop, including the fact that there has never been a leakage crack.

2. Pre-service and in-service inspections performed on piping for the Cook Plant minimize the possibility of flaws existing in such piping. The application of advanced fracture mechanics has demonstrated that if such flaws exist, they will not grow to a leakage crack when subjected to the worst case loading condition over the life of the plant.

3. If a large through-wall flaw is postulated, large margins against unstable crack extension exist for the stainless steel primary coolant piping at Cook Plant, even if subjected to the safe shutdown earthquake in combination with the loads associated with normal operation.

The application of advanced fracture mechanics technology has demonstrated that small flaws or leakage cracks (postulated or real) will remain stable. These flaws or cracks will be detected either by in-service inspection or by leakage monitoring systems long before they can grow to critical sizes. Otherwise, this could lead to large break areas such as the double-ended rupture of the largest pipe of the reactor coolant system. To date, use of this advanced fracture mechanics technology has been limited because of the definition of a LOCA in 10 CFR 50, Appendix A, which includes postulated double-ended ruptures of piping, regardless of the probability and the fact that there is no mechanistic scenario under which this event will occur. Application of the LOCA definition, without regard to this advanced

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fracture mechanics technology to large-diameter, thick-walled piping such as the primary coolant pipes of a PWR, imposes a severe penalty in terms of backfit cost and occupational radiation exposure. Massive pipe whip restraints which would be required without the application of fracture mechanics technology must be installed and then removed for in-service inspections. As documented in the NRC's Value-Impact Statement for Generic Issue A-2, this penalty is unreasonable because these pipes do not have a history of failing or cracking and are conservatively designed. Accordingly, for design purposes associated with protection against dynamic effects, we request that postulated pipe breaks in the reactor coolant system primary loop be eliminated from the structural design bases where established by appropriate analysis. This request does not extend to specifying design bases for containment, the emergency core cooling system, or environmental effects.

The use of advanced fracture mechanics would permit a deterministic evaluation of the stability of postulated flaws or leakage cracks in piping as an alternative to the current mandate of overly conservative postulations of piping ruptures. This request is consistent with the provisions of footnote 1 to 10 CFR 50, Appendix A, which contemplates the development of "further details relating to the type, size and orientation of postulated breaks in specific components of the reactor coolant pressure boundary."

As support for this request, in addition to the two Westinghouse topical reports referred to above, we request consideration of the following:

1. Memorandum from Darrell G. Eisenhut (NRC) to All Operating PWR Licensees, Construction Permit Holders, and Applicants for Construction Permits dated February 1, 1984 - Subject: Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04).

2. CRGR resolution of Generic Issue A-2, September 28, 1983.

3. ACRS letter dated June 14, 1983 - Subject: "Fracture Mechanics Approach to Pipe Failure."

4. Memorandum from William J. Dircks, EDO, to ACRS dated July 29, 1983, - Subject: "Fracture Mechanics Approach to Postulated Pipe Failure."

These documents, and Westinghouse Topical Reports WCAP 9558 and WCAP 9787, provide a substantial and adequate basis for limiting postulated design basis flaws in stainless steel reactor coolant system piping.

A detailed value-impact analysis has been performed by Pacific Northwest Laboratory (PNL) to assess the relative costs of using advanced fracture mechanics techniques to justify design bases for several operating PWRs instead of modifying these plants to conform to

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piping restraint designs used in more recent plants. This analysis clearly establishes that the costs, both in dollars and radiation exposure, are greater for modifying the plants than are the monetary and radiation exposure costs due to guillotine pipe ruptures, considering the low probability of such events. Indiana & Michigan Electric Company supports the conclusions reached in this analysis.

The analysis is not specific for each of the evaluated plants, but the analysis inputs are reasonable. Estimates of occupational radiation exposure rates conservatively correspond with dose rates experienced at the Cook Plant in locations where modifications would be required. Portions of the estimates of modification costs and man-hours of occupational exposure are based on estimates from utilities with operating PWRs and thus should be realistic. It should be noted, however, that the cost estimates are no longer current and are, therefore, probably low. The estimates of guillotine pipe break frequency contained in the analysis are probably too high. The estimates are based on data which is not specific to guillotine breaks of large diameter, stainless steel, nuclear grade piping. Consequently, they overestimate the probability of reactor coolant system double-ended pipe ruptures. All of these factors lead to the conclusion that the PNL analysis result is correct, but that the analysis understates the relative value of using deterministic techniques to define design bases for the affected plants. The value-impact analysis clearly establishes that advanced fracture mechanics analysis is an acceptable alternative to designing and installing plant modifications to mitigate the consequences of unrealistically postulated doubled-ended guillotine breaks.

It is unclear whether the use of advanced fracture mechanics is already permitted by 10 CFR 50, Appendix A, to define LOCA pipe break sizes. It is also unclear whether 10 CFR 50.34 and Appendix A apply to plants for which construction permits were already issued at the time these requirements for construction permit applications were issued. Nevertheless, as authorized by Generic Letter No. 84-04, Indiana & Michigan Electric Company hereby applies, pursuant to 10 CFR 50.12(a), for an exemption from the provisions of 10 CFR 50, Appendix A, authorizing alternative pipe break analyses to establish the structural design bases resulting from pipe breaks in connection with license Nos. DPR-58 and DPR-74. Further, pursuant to 10 CFR 50.12(a), we believe the requested exemption will not endanger life, property, or the common defense and security, and is in the public interest.