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FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
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SUBJECT: Discusses Westinghouse evaluation of reactor vessel supports,ECCS piping sys & component supports for breaks inside & outside reactor cavity.Probability of pipe break small.Continued reactor operation justified.

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INDIANA & MICHIGAN ELECTRIC COMPANY

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February 15, 1980
AEP:NRC:00137B

Donald C. Cook Nuclear Plants Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

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

Reference: (1) Letter No. AEP:NRC:00137 dated September 26, 1979

(2) Letter No. AEP:NRC:00137A dated December 7, 1979

(3) Westinghouse Letter No. NS-TMA-2200 dated February 6, 1980 to Mr. D.G.Eisenhut, Director DOR, NRC.

(4) Westinghouse Letter No. NS-TMA-2206 dated February 14, 1980 to Mr. D. G.Eisenhut, Director DOR, NRC.

Dear Mr. Denton:

This letter is in reference to Condition C.3 (a) to the operating License No. DPR-74 for Unit 2 of the Donald C. Cook Nuclear Plant. The license condition requires that an analytical evaluation be made of the effects of certain postulated break loads on the reactor coolant system and internals. Mr. V. Stello's letter dated January 25, 1978 made the above analysis requirements applicable to Unit No. 1. Within the schedule agreed upon with members of your staff and noted in Reference (2), Westinghouse has completed the evaluation of Reactor Vessel Supports, attached ECCS piping systems and component supports for breaks inside and outside the reactor cavity.

The completed evaluation program consists of three phases, A, B, and C.

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Phase 'A' included data acquisition from the utilities, and review of structural and hydraulic parameters for potential grouping among generically similar plants. This phase was completed on July 21, 1979. Phase 'B' (Pipe Breaks Outside Reactor Cavity) consists of evaluation of structural integrity of the NSSS component supports for breaks outside the reactor cavity and development of specific plant qualification programs as required. Phase "B" also included work required as input for reactor vessel evaluations to be performed in Phase "C" and initiation of mechanistic pipe break analyses.

Westinghouse has completed the above evaluations and issued the following detailed reports:

1) "Westinghouse Owners' Group Asymmetric LOCA Loads Evaluation"

   WCAP 9628 (proprietary)
   WCAP 9662 (non proprietary)

2) "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through -Wall Crack."

   WCAP 9559 (Proprietary)
   WCAP 9570 (Non-Proprietary)

These reports were submitted to the staff by Westinghouse Electric Corporation on February 6, 1980,Reference (3).

For the Donald C. Cook Nuclear Plant Units 1 and 2, the phase 'B' analysis concluded that the maximum stresses in the components and their supports were within the allowable stress limits and thereby the structural and operational integrity of the system components is assured and no modifications to the existing design are required.

The mechanistic pipe break study of Phase "B" involved an analysis and test program to understand the fracture mechanics characteristics of a pipe break. Both aspects of this effort are now complete. This study showed that a through-wall thickness flaw existing in the pipe material (which is easily detectable by ultrasonic testing) will have a limited crack growth under combined operating plus earthquake (SSE) loads; crack propagation will not result in a full size pipe break; the crack will remain stable. Analytical studies were also made to evaluate the effect of a suddenly appearing crack in the pipe. This analysis was done for both wrought and cast materials. In both cases the effect was minimal and there exists no danger of producing a double-ended circumferential break.

Another independent study in the fracture mechanics analysis done by NSAC/EPRI reached similar conclusions. This report in the form of a Technical Memorandum was submitted to the NRC on October 19, 1979 (in a letter to you from John E. Ward, Chairman of the AIF Committee on Reactor Licensing and Safety). This study has determined by diverse, independent analyses and experimental results, that the probability of high energy line breaks in reactor piping systems, built from either austenitic or ferritic steels is extremely small. In addition, the consequences of unanticipated, slow crack growth due to fatigue, corrosion fatigue, or stress corrosion cracking is likely to be a relatively small leakage. The analyses specifically determined that very large cracks are required to initiate ductile fracture in nuclear piping under normal loadings; if ductile fracture does initiate due to a severe overload, unstable crack extension is unlikely to occur; and the opening-up of through wall cracks are small.

These results support the conclusion that a double-ended guillotine break in the primary system piping, without any prior indication of substantial leakage, is unrealistic and need not be considered as a basis for plant design or modification and that a well established inservice inspection program, is a sufficient method to detect/prevent the development of large cracks of concern.

Phase 'C' (pipe breaks inside reactor cavity) involved the evaluation of the structural integrity of the NSSS components and supports, ECCS piping, fuel, internals and the CRDM, for breaks near the reactor vessel inlet nozzles. Westinghouse has completed a detailed evaluation of the NSSS components and their supports and a qualitative evaluation of the ECCS piping system. The results of this evaluation are being submitted to the NRC by Westinghouse Electric Corporation in Reference (4). The detailed evaluation of the remainder of the systems will be submitted by July 1, 1980 as agreed upon earlier with the Staff.

For Donald C. Cook Nuclear Plants Units 1 and 2, the analysis of the Reactor Vessel, Steam Generators, Reactor Coolant Pumps and of their supports shows that the structural and operational integrity is assured for the asymmetric LOCA loads due to a limited pipe break at the reactor vessel inlet nozzles; i.e. a break area of about one square foot. A qualitative evaluation of the ECCS piping assures that the system integrity is maintained during a LOCA event and capability to deliver ECCS flow to the core is assured. To achieve the acceptable loadings on the component supports, for the above postulated, hypothetical breaks, Westinghouse assumed break-limiting devices in the mathematical model. The details of these devices have to be finalized.

The Staff has stated that the asymmetric LOCA loads must be combined with seismic loads. We feel that this position is not justified as demonstrated by the mechanistic break analysis and testing. Furthermore, we know of no acceptable analytical technique for combining the existing elastic seismic loads with the inelastic asymmetric loads which were developed for some specific break locations. If the Staff insists upon load combination, we will work to develop an acceptable analytical technique. The design of modifications must be postponed until such a technique is developed.

Should the Staff require that the plant be modified to meet the analysis assumptions, despite the evidence presented in the mechanistic break report, the modifications could be installed during the second refueling outage after agreement is reached on the full resolution of the asymmetric loads issue. After the final resolution and approval by the NRC, two refueling cycles are required to install the modifications because detailed measurements must be made of the reactor cavity shield wall/reactor coolant pipe annulus prior to fabrication of break-limiting devices. Upon acceptance by the NRC of the analyses and acceptance of the modifications as a complete resolution of the asymmetric loads issue, Indiana & Michigan Electric Company will agree to design and install the modifications. Not until a full resolution of the issue is agreed upon will commitment of expenditure of funds be acceptable.

We agree with the NRC Staff assessment "that the probability of a pipe break resulting in substantial transient loads on the vessel support system or other structures is acceptably small because (1) the break of primary concern must be very large, (2) it must occur at a specific location, (3) the break must occur essentially instantaneously, and (4) the welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI". Therefore continued reactor operation is justified while this matter is being resolved.

Very truly yours,

R. S. Hunter
Vice President

RSH:em

cc: R. C. Callen
G. Charnoff
J. E. Dolan
R. W. Jurgensen
D. V. Shaller -Bridgman