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SCHWENCER,A. Operating Reactors Branch 1

SUBJECT: Submits interim rept re asymmetric LOCA loads study. Adresses structural integrity of reacotr vessel & supports steam generator & reacotr coolant pump supports & attached ECCS related piping.Considers no plant mods necessary.

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Carolina Power & Light Company

February 18, 1980

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Office of Nuclear Reactor Regulation
ATTENTION: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-261
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INTERIM REPORT ON ASYMMETRIC LOCA LOADS STUDY

Dear Mr. Schwencer:

In accordance with the NRC staff's request at the November 28, 1979 meeting concerning Asymmetric LOCA loads during hypothetical primary coolant pipe severences, we are submitting this interim report for H. B. Robinson Unit No. 2 which addresses: the structural integrity of the reactor vessel and supports, the steam generator and reactor coolant pump supports, and attached ECCS related piping.

Mr. Victor Stello's letter of January 25, 1978, requested that we evaluate asymmetric loads developed during a Loss of Coolant Accident (LOCA) and provide a schedule for completion of the evaluation. A Westinghouse Owners Group was formed to complete an evaluation of the type requested and developed a two-year schedule for the program. The evaluation program was divided into three phases, A, B, and C.

Phase A included data acquisition from the Utilities, and review of structural and hydraulic parameters for potential grouping among generically similar plants.

Phases B and C separated the evaluations for breaks postulated outside the reactor cavity and inside the reactor cavity. Phase B involved the actual structural assessments of plant groups and development of specific plant qualification programs as required for breaks outside the reactor cavity area. Phase C included evaluation of breaks inside the reactor cavity annulus and verification of the structural integrity of the reactor vessel and supports, reactor internal structures, fuel, and ECCS piping attached to the reactor coolant system. The integrity of the CRDM's and primary equipment supports which may be controlled by these vessel nozzle breaks is also considered in Phase C.

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Concurrent with the Phase B and C work, mechanistic pipe break analyses were also undertaken to determine if large through-wall cracks in reactor coolant system piping would propagate to a large LOCA. Results of this work have previously been submitted by Westinghouse for the Owners Group in the form of WCAP-9558, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack." This report and the NSAC/EPRI Technical Memorandum submitted to the NRC on October 19, 1979, in a letter from John E. Ward (Chairman, AIF Committee on Reactor Licensing and Safety) to Harold R. Denton, have determined, by diverse and independent analyses and experimental results, that the probability of high energy line breaks in reactor piping systems, both austenitic and ferritic, is extremely small. 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 openings of through-wall cracks are small. Therefore, the consequences of unanticipated, slow crack growth due to fatigue, corrosion fatigue, or stress corrosion cracking is likely to be relatively slow leakage.

These results support the conclusion that a double-ended guillotine break in a reactor system pipe without any prior indication of substantial leakage is unrealistic and need not be considered as a basis for plant design or modification.

Nevertheless, Phase B and Phase C asymmetric loads analyses have been continued. Results have been and will be submitted as described below:

Phase B - Evaluation of postulated reactor coolant piping breaks outside the reactor cavity.

1. Westinghouse Owners Group Report "Phase B5: Subcompartment Asymmetric Pressure Loads"; authored by D. S. Nixdorf - submitted February, 1979.

2. Structural evaluation of steam generator and reactor coolant pumps supports: WCAP-9628, "Westinghouse Owners Group Asymmetric LOCA Loads Evaluation - Phase B"; submitted February 6, 1980.

Phase C - Evaluation of postulated reactor coolant piping breaks inside the reactor cavity.

1. The Westinghouse Owners Group analyzed the "representative" plants and will present the results to the NRC in a meeting on February 21, 1979.

2. Integrity verification of the reactor vessel supports and the ECCS related piping attached to the reactor coolant system: "Interim Report for Asymmetric LOCA Loads Evaluation Phase C; Reactor Nozzle Postulated Breaks" - to be submitted by Westinghouse for the Owners Group by February 15, 1980.

3. Integrity verification of the fuel, reactor internals and the CRDM's: In accordance with an agreement with the NRC staff at the November 28, 1979 meeting, these evaluation results will be submitted by July 1, 1980.

The analytical results discussed above have been compiled because of the NRC staff's expressed desire to gain a better understanding of the asymmetric loads issue. We continue to believe that the additional mechanistic break work which the Westinghouse Owners Group performed presents sufficient justification to eliminate double-ended guillotine breaks as a basis for plant design. We urge that review of the mechanistic break topical report, WCAP-9558, be continued and that its conclusions be adopted as a basis for the resolution of this issue.

For the above reasons, we believe that no plant modifications are necessary to maintain public health and safety. However, should the staff require modifications be installed, the actual installation of the modifications could not occur until the second refueling outage after an agreement is reached on full resolution of this issue. Two refueling outages would be required to install any modifications because detailed measurements would be required of the reactor cavity shield wall/reactor coolant pipe annulus prior to fabrication of break limiting devices. The NRC staff has stated that asymmetric LOCA loads must be combined with seismic loads. We feel that this position is not justified as demonstrated in the mechanistic break work; however, seismic load tables will be submitted to the staff for the Owners Group by Westinghouse at the next NRC meeting, tentatively set for February 26, 1980.

We agree with, and our analyses support as conservative, 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 Section XI". Therefore, continued reactor operation is justified while this matter is being resolved.

Yours very truly,

E. E. Utley

E. E. Utley
Executive Vice President
Power Supply and Customer Services

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