REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8002200449 DOC.DATE: 80/02/15 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
STN-50-485 Sterling Power Project Nuclear, Unit 1, Rochester 05000485
AUTH.NAME AUTHOR AFFILIATION
WHITE,L.D. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
Office of Nuclear Reactor Regulation
ZIEMANN,D.L. Operating Reactors Branch 2

SUBJECT: Responds to V Stello 780125 request for evaluation of assysmmetric loads developed during LOCA & schedule for completion.Provides status of evaluation program & results of analyses completed.

DISTRIBUTION CODE: A030S COPIES RECEIVED:LTR 1 ENCL 0 SIZE: 3
TITLE: Asymmetric Loca Loads

NOTES: 1 CY: C. HOFMAYER, J. SHAPAKER

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	05 BC	5				
INTERNAL:	01 REG FILE	1		02 NRC PDR	1	
	10 HOSFORD,S	1		12 TA/EDO	1	
	13 I&E	2		15 OELD	1	
	17 REAC SFTY BR	1		18 CORE PERF BR	1	
	19 ENGR BR	1		20 PLANT SYS BR	1	
	21 MECH ENGR BR	1		22 CONTMNT SYS	1	
	23 ANALYS BR	1		HANAUER,S.	1	
	STS GROUP LEADR	1				
EXTERNAL:	03 LPDR	1		04 NSIC	1	
	24 ACRS	16	16			

STANDARDIZED PLANT
I & E 3 CYS OF ALL MATL

FEB 25 1980

T

MA
4

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~39~~ 44 ENCL ~~39~~ 0

[illegible]

[illegible]
[illegible]

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

February 15, 1980

Director of Nuclear Reactor Regulation
Attn: Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch #2
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Ziemann:

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 purpose of this letter is to review the status of the evaluation program and to provide results of the analyses completed.

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.

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 (Westinghouse letter NS-TMA-2200 dated February 6, 1980) 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

A030
S,1/0

8002200 449

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 demonstrated 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 consequence 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. Westinghouse Owners' Group report "Phase B5: Subcompartment Asymmetric Pressure Loads" authored by D. S. Nixdorf was presented to the NRC Staff in February, 1979. The remainder of the Phase B work covering steam generator and reactor coolant pump integrity and supports evaluation, is reported in WCAP 9628, Westinghouse Owners' Group Asymmetric LOCA Loads Evaluation Phase B, which was submitted February 6, 1980 by Westinghouse letter NS-TMA-2200 to Mr. Darrel Eisenhut. Phase C results for verification of the structural integrity of the reactor vessel and supports and ECCS piping attached to the reactor coolant system are being submitted for the Owners Group under separate cover by Westinghouse letter NS-TMA-2206 to Mr. Darrel Eisenhut dated February 14, 1980. In accordance with an agreement reached with the NRC Staff in November, 1979 results of evaluations of reactor internal structures, fuel, and control rod drive mechanisms will be provided by July, 1980. In addition, the Westinghouse Owners' Group also analyzed two "representative" plants and presented the results to the NRC in a meeting on February 21, 1979. The "representative" plant analyses used nominal existing plant configurations. The specific plant analysis for R. E. Ginna assumed a modified plant configuration, including for example, break limiting devices in the reactor cavity shield wall.

The above analysis results 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 undertook 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.

We do not believe that backfitting the R.E. Ginna plant will provide substantial, additional protection for the public health and safety. To the contrary, modifications will impose additional costs upon our customers and additional radiation exposure upon those installing the modifications with no measurable benefit. However, should the staff require under Section 50.109 of the Code of Federal Regulations, 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 an agreement is reached on a full resolution of the asymmetric loads issue. 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 by the mechanistic break work. If the staff insists upon seismic and LOCA load combination, the design of modifications must be postponed until after completion of the Systematic Evaluation Program (SEP) seismic review and should be coordinated with potential SEP modifications. 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.

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 Code Section XI" (Enclosure 1 to Victor Stello's January 25, 1978 letter to all PWR licenses). Therefore, continued reactor operation is justified while this matter is being resolved.

Sincerely yours,

L D White, Jr.

L. D. White, Jr.

LDW:np