

MAY 12 1988

Docket No. 50-348

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291-0400

Dear Mr. McDonald:

SUBJECT: APPROVAL OF FRACTURE MECHANICS ANALYSIS OF TWO REACTOR VESSEL
FLAW INDICATIONS IN LOOP B, HOT LEG NOZZLE TO SHELL WELD -
JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. 67960)

By letters dated April 28 and May 5, 1988, you provided results of a recent Inservice Inspection (ISI) examination for Unit 1. During the eighth refueling outage, ISI examinations of certain reactor vessel welds and ligaments were performed in accordance with the ASME Code, Section XI. The entire examination was completed on April 24, 1988.

During the scheduled ISI of the reactor vessel, four flaw indications were detected that exceed the acceptance standards of ASME Code, Section XI. To provide enhanced characterization and sizing of these indications, the Ultrasonic Data Recording and Processing System (UDRPS) was used. The UDRPS data revealed two indications in the lower shell longitudinal seam weld within the cladding. Also, two flaw indications located in the loop B hot leg nozzle-to-shell weld exceed the standards of ASME Code, Section XI. Therefore, the component would be unacceptable for service unless such flaws are removed or repaired. However, as allowed by ASME Code, a fracture mechanics evaluation, needed to demonstrate acceptance for service by evaluation and reexamination, was provided for our approval. We have completed our review.

Our enclosed safety evaluation concludes the following:

1. The fracture mechanics analysis demonstrates that the two flaw indications in the outlet nozzle-to-shell weld will not grow during the life of the plant to a size that will affect the integrity of the reactor vessel.
2. These flaw indications are conditionally acceptable for service. Therefore, augmented inservice inspection is required during the next three inspection periods pursuant to ASME Code, Section XI, paragraph IWB-2420(b).

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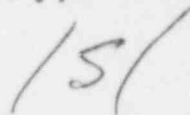
Docket File (NRC PDR + LOCAL PDR)
PD21 r/f ACRS (10)
S. Varga (14E4)
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E. Reeves(2)
OGC
E. Jordan (MNBB 3302)
J. Partlow (9A2)
B. Elliot (9H15)
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PDR

- 3. You responded to Generic Letter 83-15 by letter dated October 26, 1983 committing to the implementation of Regulatory Guide 1.150. Thus, in the event that future examinations of the reactor vessel detect flaw indications that extend from the inside diameter past the clad to the base metal interface, you must address the effects of the cladding in all required fracture mechanics evaluations.
- 4. The 1st and 2nd interval examinations on the reactor vessel were performed during the same refueling outage. Paragraphs 10 CFR 50.55a(g)(4)(i) and (ii) do not permit inservice inspections that are performed during two intervals to coincide because the regulation addresses the requirement for successive intervals. Region II will review this item during the review of the 90-day ISI report.

Please contact us should you have any questions.

Sincerely,



Edward A. Reeves, Sr. Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

OFC	:LA:PD21:DRPR:PM:PD21:DRPR:D:PD21:DRPR	:	:	:	:
NAME	:PAnderson:ch:EReeves	:	:EAdensam	:	:
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