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SUBJECT: Provides rev to original relief request for augmented reactor vessel insp.

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INDIANA
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May 6, 1996

AEP:NRC:0969AP
10 CFR 50.55a

Docket Nos.: 50-315

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Donald C. Cook Nuclear Plant Unit 1
REQUEST FOR RELIEF FOR AUGMENTED REACTOR
VESSEL IN-SERVICE INSPECTION

References

(1) Letter AEP:NRC:0969AI, "Donald C. Cook Nuclear Plant Units 1 and 2, Request for Relief for Augmented Reactor Vessel In-service Inspection," dated July 28, 1995.

(2) Letter AEP:NRC:0969AO, "Donald C. Cook Nuclear Plant Units 1 and 2, Request for Relief for Augmented Reactor Vessel In-service Inspection, Additional Information," dated April 30, 1996

The purpose of this letter is to provide a revision to our original relief request for the augmented reactor vessel inspection. Our original request, which was submitted in Reference 1, requires revision as a result of the inspection which was conducted during the 1995 unit 1 refueling outage.

Our original request was based on estimates of the percentage of each weld that could be examined. Following the examination of unit 1, we now have data which provides the actual percentage coverage of each weld. The data show that the coverage for six welds is less than the 90% coverage required by 10 CFR 50.55a(g)(6)(ii)(A)(2). As was noted in Reference 2, which

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provided additional information regarding our original relief request, three additional welds were found during the unit 1 examination, to have less than 90% coverage. These welds are included in this revised relief request which supercedes the relief request submitted in Reference 1.

Sincerely,

EE Fitzpatrick

E. E. Fitzpatrick
Vice President

Attachment

cc: A. A. Blind
G. Charnoff
H. J. Miller
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

Attachment to AEP:NRC:0969AP

Background Information and Justification

10 CFR 50.55 Code Relief

For the Augmented Reactor Pressure Vessel Shell Welds Examination

for Cook Nuclear Plant Unit 1

Background for Augmented Vessel Examination Code Relief Request

I Code Relief Request

Code relief is requested for the following unit 1 reactor pressure vessel (RPV) shell welds which were found to have less than 90% coverage during the inspection conducted during the 1995 refueling outage.

Category I.D.	Item #	Component description
Unit 1		
B-A	B1.11	Circumferential weld (lower head to lower shell weld)
B-A	B1.12	Longitudinal shell weld (upper shell at 26.5 degrees)
B-A	B1.12	Longitudinal shell weld (upper shell at 146.5 degrees)
B-A	B1.12	Longitudinal shell weld (lower shell at 60 degrees)
B-A	B1.12	Longitudinal shell weld (lower shell at 180 degrees)
B-A	B1.12	Longitudinal shell weld (lower shell at 300 degrees)

II Code Requirements

ASME Section XI, 1983 Edition Summer Addendum, Table IWB-2500-1, Category B-A, Item B1.10 requires volumetric examination of the beltline region of the RPV shell welds for each ten year interval following the first ten year interval. 10 CFR 50.55a(g)(6)(ii)(A) requires that an augmented reactor vessel weld examination be conducted prior to the end of the current interval. 10 CFR 50.55a further states that essentially 100% of the weld length (no less than 90%) is to be examined and if a determination is made that the licensee is unable to satisfy these requirements, information shall be submitted to the commission to support the determination and a proposed alternative shall be made that would provide an acceptable level of quality and safety.

III Basis for code relief

Table 1 identifies the welds for which relief is requested and indicates the examination coverage percentages obtained during the unit 1 examination.

Reactor pressure vessel shell welds are examined from the inside surface using automated ultrasonic equipment. The examination of the shell to lower head weld is limited to less than 90% due to the position of the core support lugs which provide an anti-rotation feature for the core barrel. These core support lugs inhibit the equipment access required to perform a code ultrasonic (UT) exam of the shell weld from both sides of the weld.

Two of the longitudinal upper shell welds and all of the longitudinal lower shell welds could not be examined at coverage percentages of 90% or better due to physical and geometric interferences (See Table 1).

The automated RPV examinations were performed with modified equipment and tooling designed to optimize coverage. Automated equipment set-up was also optimized (indexed as close to the obstructions as possible) to afford maximum coverage.

IV Proposed Alternatives

As stated above, the automated RPV examinations were performed with modified equipment and tooling designed to optimize coverage. Automated equipment set-up was also optimized (indexed as close to the obstructions as possible) to afford maximum coverage. Additionally, the possibility of examining from the RPV outside surface was reviewed for improved coverage where limitations resulted in coverages less than 90%. This review is further detailed in the following justification. As an alternative, we are proposing that the examination coverage obtained for these six (6) welds be considered to provide an acceptable level of quality and safety.

V Justification for Granting of Code Relief

Examination of 100 percent of RPV shell welds would result in undue hardship and burden with no commensurate safety benefit realized. Examination of the accessible weld volume provides sufficient and reasonable assurance of vessel integrity. This reduction in the expected examination coverage will not endanger life or property or the common defense and security because the reactor coolant system is designed and constructed to have a low probability of gross rupture or significant leakage throughout its design life and technical specification 3.4.6.2 places limits on the amount of reactor coolant system leakage during operation. The most likely weld failure would be a crack which would allow additional coolant to leak from the system. Any such leakage would be detected and retained within the containment building. Should this occur, and leakage exceeds the technical specification allowables, the appropriate action statement would be followed. Additionally, past examinations of the accessible RPV shell welds have revealed no recordable indications and it is reasonable to conclude the same results for these inaccessible welds would be obtained.

We have reviewed the possibility of performing examination of the subject welds from the outside surface of the RPV. This could only be achieved by the removal of the RPV from the cavity due to the close proximity of the concrete biological shield wall with outside surface of the RPV. Additionally, if access to the outside surface could be obtained, a high radiation exposure associated with the scaffolding, insulation removal and replacement and UT examination, is predicted. We therefore believe that examination from the RPV outside surface would cause significant undue hardship and burden with no commensurate safety benefit realized.

Table 1

RPV shell weld examination actual coverages (less than 90%) based on Cook Nuclear Plant Unit 1 1995 examination

Unit	Weld Number	Exam Area Identification	Actual Coverage (%)	Comments
1	RPV-D	Lower head to lower shell.	69	Limitation due to core support anti-rotation lugs.
1	RPV-VA1	Upper shell at 26.5'.	80	Limitation due to intersecting nozzle.
1	RPV-VA2	Upper shell at 146.5'.	62	Limitation due to intersecting nozzle.
1	RPV-VC1	Lower shell at 60°	80	Limitation due to location of core support lugs
1	RPV-VC2	Lower shell at 180°	80	Limitation due to location of core support lugs
1	RPV-VC3	Lower shell at 300°	80	Limitation due to location of core support lugs