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SUBJECT: Purposes to revise original relief request for augmented reactor vessel in-service insp.

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AEP
INDIANA
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July 29, 1996

AEP:NRC:0969AR
10 CFR 50.55a

Docket Nos.: 50-316

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

Gentlemen:

Donald C. Cook Nuclear Plant Unit 2
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:0969AP, "Donald C. Cook Nuclear Plant Unit 1, Request for Relief for Augmented Reactor Vessel In-service Inspection, Additional Information," dated May 6, 1996.

The purpose of this letter is to revise our original relief request for the augmented reactor vessel inspection. Our original request (Reference 1) requires revision as a result of the inspection that was conducted during the 1996 unit 2 refueling outage.

Our original request for both unit 1 and unit 2 was based on estimates of the percentage of each weld that could be examined. Unit 1's relief request was modified (Reference 2) following the determination of the actual examination coverage for the reactor vessel welds. Following the recent examination of unit 2, we now have the actual percentage coverage of each weld. The coverage for four reactor pressure vessel shell welds does not exceed the 90% coverage required by 10 CFR 50.55a(g)(6)(ii)(A)(2).

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We request that code relief be granted for these four welds as every effort was made to maximize the coverage, including the application of ASME Section XI, Appendix VIII techniques. This revised relief request supersedes the relief request submitted in Reference 1 for unit 2 of the Donald C. Cook Nuclear Plant.

Sincerely,

E. E. Fitzpatrick

E. E. Fitzpatrick
Vice President

llg

Attachment

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DC-N-6015.1

Attachment to AEP:NRC:0969AR

Background Information and Justification

10 CFR 50.55a Code Relief

For the Augmented Reactor Pressure Vessel Shell

Welds Examination for Cook Nuclear Plant Unit 2

Background for Augmented Vessel Examination Code Relief Request

I Code Relief Request

Code relief is requested for four unit 2 reactor pressure vessel (RPV) shell welds that were found to have less than 90% coverage during the inspection conducted during the 1996 refueling outage.

Table 1 identifies the welds for which relief is requested and indicates the examination coverage percentages obtained during the unit 2 examination.

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 (greater than 90%) is to be examined and that, 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

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 that provide an anti-rotation feature for the core barrel. These core support lugs inhibit the equipment access required to perform a 100% code ultrasonic (UT) exam of the shell weld from both sides of the weld.

The three longitudinal upper shell welds could not be examined at coverage percentages of 90% or better due to physical and geometric interferences associated with the nozzle integral extensions (See Table 1).

The automated RPV examinations were performed with modified equipment and tooling designed to accommodate the maximum coverage possible. Automated equipment set-up was also optimized (indexed as close to the obstructions as possible) to afford maximum coverage. Additionally, paragraph IWA-2240 of ASME Section XI was invoked to apply performance demonstration initiative (PDI) techniques (ASME Section XI, Appendix VIII), as amended by the PDI program description (Reference 1), for the purpose of extending coverage of these welds.

Proposed Alternatives

As an alternative to the greater than 90% requirement for this inspection, we are proposing that the examination coverage obtained on these welds be considered to provide an acceptable level of quality and safety.

V Justification for Granting of Code Relief

Examination of 100% 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 that 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 allowable, 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.

An alternative examination was conducted on all shell welds by invoking paragraph IWA-2240 of ASME Section XI that allows the use of an alternative examination if it is demonstrated to the authorized nuclear inspector that the results are equivalent or superior to the code specified method. Appendix VIII of the ASME Section XI code, 1992 edition, as amended by the PDI program description, was used for all B1.10 shell welds and its use extended the coverage where limitations existed. The use of Appendix VIII techniques has increased the quality of examination of these RPV shell welds compared to the conventional (ASME Section V) method of qualification due to the demonstration of the procedures and the capabilities of equipment and personnel on full sized test blocks using flaws that are typical of the planar flaws expected in reactor pressure vessel welds. We have reasonable assurance that the Appendix VIII methods are superior in terms of detecting and sizing indications compared to the conventional code qualification.

Table 2 is provided to indicate the estimated total coverage for unit 2, and is consistent with the results obtained for the unit 1 B1.10 welds (Reference 2). Approximately 94% coverage of the total weld length was obtained for those welds that are subject to this augmented examination. This compares with a coverage of 88.8% for unit 1 that does not include the benefits of ASME Section XI, Appendix VIII techniques.

We have reviewed the possibility of performing the 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 to the outside surface of the RPV. Additionally, even 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 believe that the examination from the RPV outside surface would cause significant undue hardship and burden with no commensurate safety benefit.

VI References

1. Performance Demonstration Initiative (PDI) Program Description, Revision 0, dated September 14, 1994.

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 May 1, 1996.

Table 1

RPV shell weld examination actual coverages which are 90% or less based on Cook Nuclear Plant Unit 2 1996 examination

Weld Number	Exam Area Identification	Actual Coverage (%)	Comments
RPV-D	Lower head to lower shell.	80	Limitation due to core support anti-rotation lugs
RPV-VA1	Upper shell at 22°.	90	Limitation due to proximity of outlet nozzle.
RPV-VA2	Upper shell at 113°.	90	Limitation due to proximity of inlet nozzle.
RPV-VA3	Lower shell at 247°	90	Limitation due to proximity of inlet nozzle

Table 2

Cook Nuclear Plant Unit 2 Reactor Vessel Shell Weld Limitations

Weld	Weld Number	Length, Feet	Coverage, %	Length Covered, Feet	Total Coverage %
Circumferential Welds					
Middle to upper	RPV-B	45.27	100	45.27	
Lower to middle	RPV-C	45.27	100	45.27	
Lower to lower head	RPV-D	45.27	80	36.21	
Total		135.81		126.75	93.33
Longitudinal Welds					
Upper shell @ 22 deg	RPV-VA1	8.08	90	7.28	
Upper shell @ 113 deg	RPV-VA2	8.08	90	7.28	
Upper shell @ 247 deg	RPV-VA3	8.08	90	7.28	
Middle shell @ 170 deg	RPV-VB1	9.00	100	9.00	
Middle shell @ 350 deg	RPV-VB2	9.00	100	9.00	
Lower shell @ 90 deg	RPV-VC1	9.00	100	9.00	
Lower shell @ 270 deg	RPV-VC2	9.00	100	9.00	
Total		60.25		57.83	95.98
Grand Total		196.06		184.58	94.15