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RECIP. NAME RECIPIENT AFFILIATION
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SUBJECT: Requests relief from requirements of 10CFR50.55 due to
inability to meet requirements specified in 10CFR50.55.
Drawings encl.

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July 28, 1995

AEP:NRC:0969AI

Docket Nos.: 50-315
50-316

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

Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
REQUEST FOR RELIEF FOR AUGMENTED REACTOR
VESSEL IN-SERVICE INSPECTION

The purpose of this letter is to request relief from the requirements of 10 CFR 50.55a(g)(6)(ii)(A)(5) (augmented reactor vessel shell weld inspection) because we have found that we are unable to meet the requirements specified in 10 CFR 50.55a(g)(6)(ii)(A). Specifically, we are unable to examine greater than 90% of seven reactor vessel welds, because of interferences created by structures within the reactor vessel (see attachment). The amount of weld accessibility for inspection on the unit 1 reactor vessel is provided in the attachment to this letter. The amount of weld accessibilities on the unit 2 reactor vessel is expected to be approximately the same as unit 1. The ASME Code, Section XI requires that essentially 100% of the weld be examined, and, for the augmented inspection, 10 CFR 50.55a(g)(6)(ii)(A)(2) defines "essentially" as greater than 90%. We are, therefore, requesting code relief for this inspection. The bases for the request are presented in the attachment to this letter. This reactor vessel weld inaccessibility condition is also being reviewed by the Nuclear Regulatory Commission for other plants, such as Ft. Calhoun.

The augmented reactor vessel weld inspection is required to be completed by the end of our second ten-year inspection interval, June 30, 1996. We intend to perform these examinations during the next refueling outages for unit 1 and unit 2, currently scheduled for September 16, 1995, and March 16, 1996, respectively.

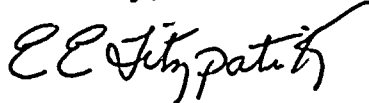
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Appl
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Approval of the relief requests is not required until the end of the respective ten year intervals, June, 1996 for both unit 1 and unit 2. An expeditious review, however, would be beneficial in view of the upcoming outages.

Sincerely,



E. E. Fitzpatrick
Vice President

plt

Attachment

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

Attachment to AEP:NRC:0969AI

Background Information and Justification

10 CFR 50.55 Code Relief

For the Augmented Reactor Pressure Vessel Shell Welds Examination
for Cook Nuclear Plant Units 1 & 2

Background for Augmented Vessel Examination Code Relief Request

I Code Relief Request

Code relief is requested for the following reactor pressure vessel (RPV) shell welds which are scheduled to be inspected during the next units 1 & 2 refueling outages which will be the final outages for the second ISI interval:

Category I.D.	Item #	Component description
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Unit 1

B-A	B1.11	Circumferential weld (lower head to shell weld)
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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)
-----	-------	--

Unit 2

B-A	B1.11	Circumferential Weld (lower head to shell weld)
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B-A	B1.12	Longitudinal shell weld (upper shell at 22 degrees)
-----	-------	---

B-A	B1.12	Longitudinal shell weld (upper shell at 113 degrees)
-----	-------	--

B-A	B1.12	Longitudinal shell weld (upper shell at 247 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 one 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 inspection 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 less than 90%, code relief must be requested in accordance with 10 CFR 50.55a(g)(6)(ii)(A)(5) with the justification for doing less than 90%.

III Basis for code relief

Figures 1 & 2 identify the RPV shell welds for Cook Nuclear Plant units 1 & 2 RPVs. Table 1 identifies the welds for which relief is requested and indicates the estimated examination coverage percentages for unit 1. The unit 2 coverage is expected to be similar to the unit 1 coverage.

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.

We also anticipate that the longitudinal upper shell welds which intersect outlet nozzles can not be examined at coverage percentages of 90% or better due to physical and geometric interferences (see Table 1).

The automated examinations on both units 1 & 2, which are scheduled for the next refueling outages, (unit 1 1995 and unit 2 1996), will be performed with modified equipment and tooling designed to accommodate the maximum coverage possible. Automated equipment set-up will also be optimized (indexed as close to the obstructions as possible) to afford maximum coverage.

Alternate examination from the outside surface of the lower shell weld to the RPV is not practical due to high levels of radiation expected for the personnel who will install and tear down scaffolding and insulation, and for the personnel who will conduct the volumetric examinations.

IV Alternate Examinations

No alternate examinations are proposed at this time. Special tooling has been designed and will be used to maximize coverage. Equipment set-up (indexing as close to the obstructions as possible) will be used during the upcoming outages.

V Justification for Granting of Code Relief

Examination of 100 percent of RPV shell welds is not practical. Examination of the accessible weld volume should be sufficient to provide reasonable assurance of vessel integrity. This change 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 reactor coolant to leak from the system. Any such leakage would be detected and retained within the containment building. Should this occur, the appropriate action statement would be followed if the leakage exceeded the technical specification allowables. 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.

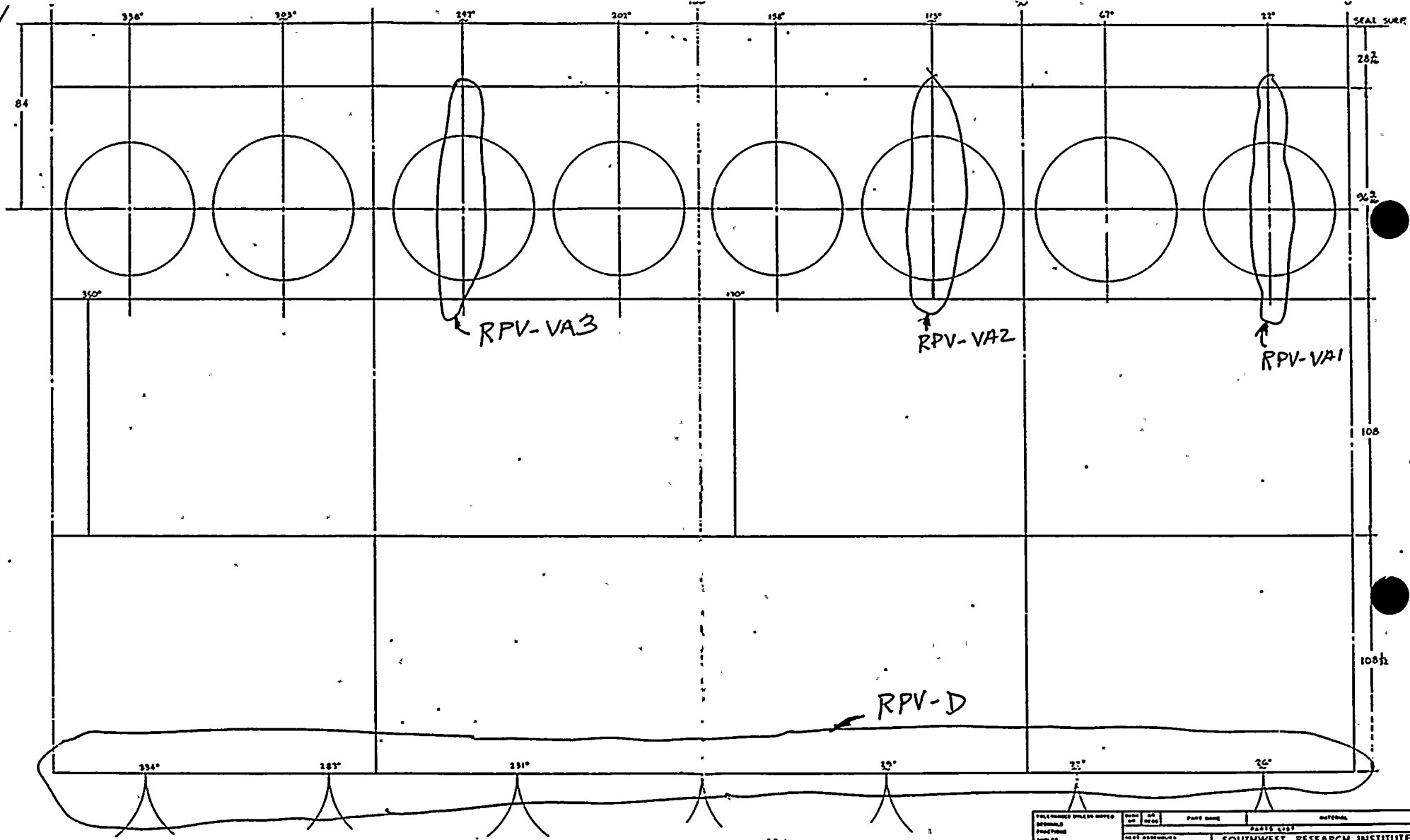
Alternate examination from the outside of the RPV is impractical due to the expected high radiation exposure associated with the scaffolding and insulation removal and replacement and UT examination with no commensurate safety benefit realized.

Table 1

Relief Request Estimated Examination Coverages
For Cook Nuclear Plant Units 1 & 2

Unit	Weld Number	Exam Area Identification	Estimated Coverage (%)	Comments
1	RPV-D	Lower shell to lower head.	76	Limitation due to core support anti-rotation lugs.
1	RPV-VA1	Upper shell at 26.5'.	85	Limitation due to intersecting nozzle.
1	RPV-VA2	Upper shell at 146.5'.	88	Limitation due to intersecting nozzle.
2	RPV-D	Lower shell to lower head.	Unknown*	Limitation due to core support anti-rotation lugs.
2	RPV-VA1	Upper shell at 22'.	Unknown*	Limitation due to intersecting nozzle.
2	RPV-VA2	Upper shell at 113'.	Unknown*	Limitation due to intersecting nozzle.
2	RPV-VA3	Upper shell at 247'.	Unknown*	Limitation due to intersecting nozzle.

* Unit 2 percentage coverage estimate is not available but is believed to be approximately 76% for the lower shell to lower head weld and 85-88% for the intersecting nozzles.



REF. DWGS.
C.B. & I. CO. CONTRACT 68-3262 DWG. 1 REV 5
DWG. 2 REV 4

FIGURE 2
Reactor Pressure Vessel
Rollout Drawing
Unit 2

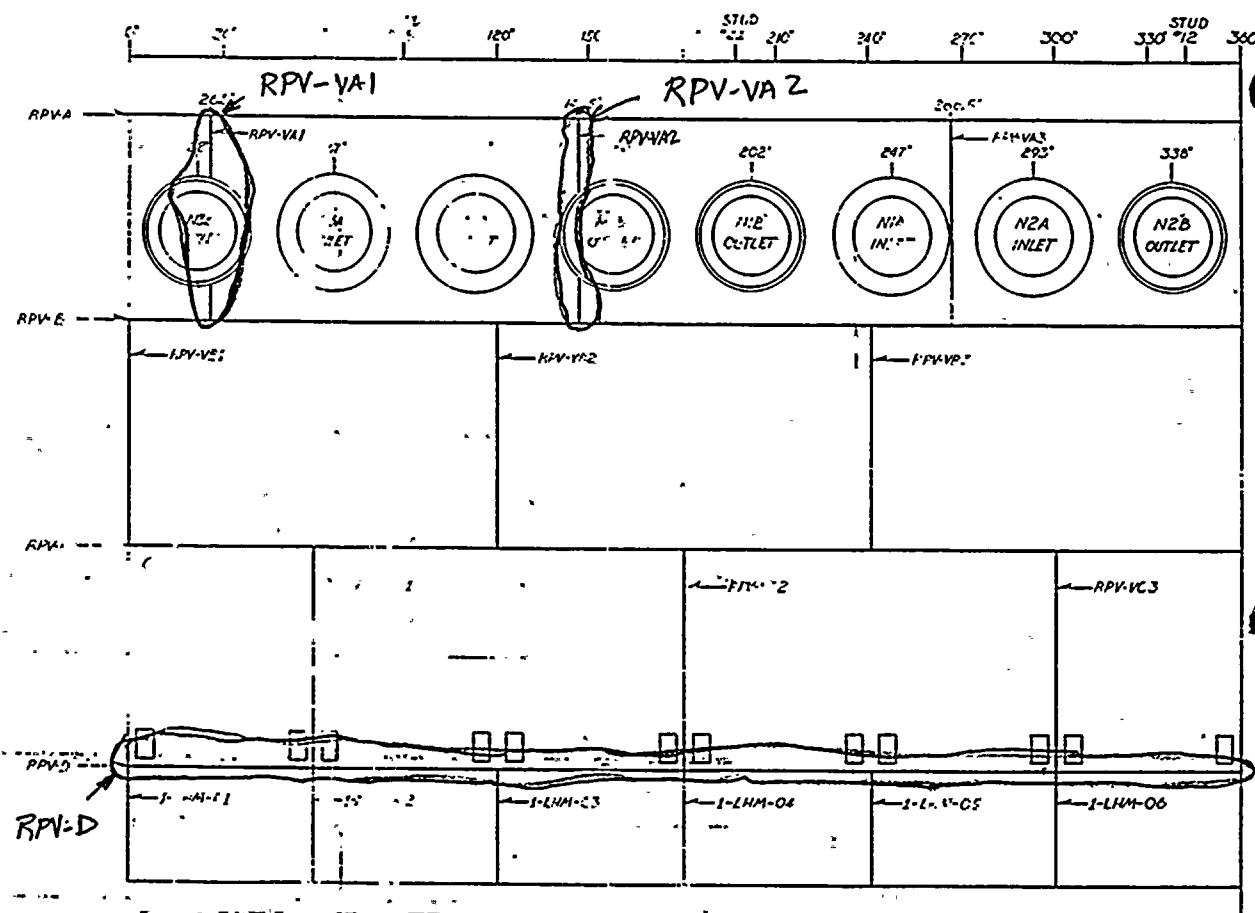
VESSEL ROLLOUT AT 192 1/2"

REV	DESIGN	DATE	BY	CHKD BY	APP'D BY	DATE

TOLERANCES UNLESS NOTED		DATE	BY	DATE	BY
APPROVED	DATE	DATE	BY	DATE	BY
DESIGNED BY	DATE	DATE	BY	DATE	BY
CHECKED BY	DATE	DATE	BY	DATE	BY
SIGNED BY		DATE	BY	DATE	BY
DRAWN BY		DATE	BY	DATE	BY
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SCALE 1/20		D-3378911			

SOUTHWEST RESEARCH INSTITUTE
QUALITY ASSURANCE SYSTEMS AND ENGINEERING DIVISION
SAN ANTONIO, TEXAS
VESSEL ROLLOUT

SHEET 1 OF 1



EXAMINATION NO			
SOUTHWEST RESEARCH INSTITUTE QUALITY ASSURANCE SYSTEMS AND ENGINEERING DIVISION SAN ANTONIO, TEXAS			
VESSEL DEVELOPMENT D.C.COOK UNIT 1			
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