UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

AUGMENTED VESSEL EXAMINATION ALTERNATIVE

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT 1

DOCKET NO. 50-315

1.0 INTRODUCTION

Section 50.55a(g)(6)(ii)(A) of 10 CFR contains requirements for an augmented examination of reactor vessels. This section requires licensees to implement an augmented examination of "essentially 100%" of the reactor pressure vessel (RPV) shell welds. The shell welds are specified in the 1989 Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB-2500-1, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," Item B1.10. This ASME category includes Item B1.11, circumferential shell welds, and Item B1.12, longitudinal shell welds. Section 50.55a(g)(6)(ii)(A)(2) of 10 CFR defines "essentially 100%" examination as "more than 90 percent of the examination volume of each weld." The schedule for implementation of the augmented inspection is dependent upon the number of months remaining in the 10-year inservice inspection (ISI) interval that was in effect on September 8, 1992. Section 50.55a(g)(6)(ii)(A)(5) allows licensees unable to completely satisfy the requirements of the augmented reactor vessel examination to propose an alternative that would provide an acceptable level of quality and safety. A licensee may use its proposed alternative when authorized by the Office of Nuclear Reactor Regulation.

In a letter dated July 28, 1995, Indiana Michigan Power Company, the licensee, submitted an alternative augmented examination regarding the RPV shell welds at D.C. Cook Nuclear Plant, Unit 1. In letters dated January 25, 1996, April 30, 1996, and May 6, 1996, the licensee deferred the alternative to the required augmented reactor vessel exam for D.C. Cook Nuclear Plant, Unit 2, and provided additional and clarifying information to support the request for Unit 1. The licensee examined the RPV internal surface to the extent practical, even though 90 percent coverage was not attained. The licensee requested authorization not to perform any additional or alternative RPV examinations to complete the requirements.

Enclosure

2.0 EVALUATION AND CONCLUSIONS

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory, has evaluated the information provided by the licensee in support of its second 10-year interval ISI augmented vessel examination alternative for D.C. Cook Nuclear Plant, Unit 1.

Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the attached Technical Letter Report. The staff concluded that the licensee has maximized examination coverage to the extent practical, and that imposing additional examinations would result in a considerable hardship without a compensating increase in the level of quality and safety.

Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the licensee's proposed alternative using the augmented RPV examination of the accessible weld volumes from the inside surface provides an acceptable level of quality and safety to satisfy the requirements of 10 CFR 50.55a(g)(6)(ii)(A).

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TECHNICAL LETTER REPORT
SECOND 10-YEAR INTERVAL ISI
AUGMENTED VESSEL EXAMINATION ALTERNATIVE
INDIANA MICHIGAN POWER COMPANY
D. C. COOK NUCLEAR PLANT, UNIT 1
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1.0 INTRODUCTION

The licensee, Indiana Michigan Power Company, submitted a proposed alternative regarding the augmented reactor vessel shell weld examinations required by 10 CFR 50.55a(g)(6)(ii)(A) for the second 10-year ISI interval at D. C. Cook Nuclear Plant, Units 1 and 2. The alternative was contained in a letter dated July 28, 1995. Additional information, which included the examination results for the Unit 1 augmented reactor vessel inspection, was included in a letter dated January 25, 1996. In a conference call between the licensee, Nuclear Regulatory Commission (NRC), and Idaho National Engineering Laboratory (INEL) staff on March 28, 1996, it was agreed that this evaluation would be performed for Unit 1 only. This was because quantitative examination coverage data for Unit 2 was not yet available. Additional information was provided in a letter dated April 30, 1996. A revised proposed alternative for Unit 1 was provided in a letter dated May 6, 1996. The INEL staff has evaluated the subject proposed alternative for Unit 1 in the following section.

2.0 EVALUATION

The Code of record for the second 10-year ISI interval at the D. C. Cook Nuclear Plant, Unit 1, is the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, 1983 Edition with the Summer 1983 Addenda. The information provided by the licensee in support of the proposed alternative has been evaluated and the basis for disposition is documented below.

10 CFR 50.55a(g)(6)(ii)(A) Augmented Examination of the Reactor Vessel

Regulatory Requirement: 10 CFR 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel", requires the examination of essentially 100% of reactor vessel shell welds specified in Items B.11 and B.12 of Examination Category B-A of the 1989 Edition of Section XI. Essentially 100% is defined as more than 90% in Paragraph 50.55a(g)(6)(ii)(A)(2).

Attachment

Licensee's Proposed Alternative: Due to interferences from core support lugs and nozzles, the licensee has proposed an alternative to the augmented reactor pressure vessel requirements for the following reactor pressure vessel (RPV) shell welds:

Weld Identification	Ultrasonic Scan Coverage
RPV-D	69%
RPV-VA-1	80%
RPV-VA-2	62%
RPV-VC-1	80%
RPV-VC-2	80%
RPV-VC-3	80%

Licensee's Basis for the Alternative (as stated):

"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 setup was also optimized (indexed as close to the obstructions as possible) to afford maximum coverage.

"Examination of 100 percent of the 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 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 that 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 the 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 examinations, 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.

Licensee's Proposed Alternative (as stated): "As stated above, the automated RPV examinations were performed with modified equipment and tooling designed to optimize coverage. Automated equipment setup was also optimized (indexed as close to the obstructions as possible) to afford maximum coverage. Additionally, the possibility of examining form the RPV outside surface was reviewed for improved coverage where limitations resulted in coverages less than 90%. This review is further detailed in the justification above. 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."

Evaluation: For compliance with the augmented reactor vessel examination requirements, the licensee must volumetrically examine essentially 100% (> 90%) of each of the Item B.11 circumferential and Item B.12 longitudinal shell welds. However, the core support lugs and nozzles in the D. C. Cook Unit 1 reactor pressure vessel limit the ultrasonic scanning access and prevent obtaining complete examination coverage of the required volume. Supplementary examinations from the external surface to increase examination coverage are not feasible because it would require removal of the RPV from the cavity due to the close proximity of the concrete biological shield wall with the outside surface of the RPV. Also, if access to the outside surface could be obtained, there would be the burden associated with high radiation exposure levels associated with scaffolding, insulation removal and replacement, and performance of the examinations. Therefore, essentially 100% coverage of the subject reactor pressure vessel welds is not achievable. To obtain complete volumetric coverage, design modifications would be required. Imposition of this requirement would cause a considerable burden on the licensee.

The volumetric examinations of the Unit 1 reactor pressure vessel shell welds have been performed to the extent practical from the inside surface using mechanized inspection equipment. RPV-D, the Item B.11 lower shell-to-lower head weld, received a 69% examination. The other Item B.11 circumferential shell welds, RPV-B (upper shell-to-middle shell) and RPV-C (middle shell-to-lower shell), each received 100% examinations. This results in an 89.7% average coverage for the circumferential shell welds.

The augmented examination of the nine Item B1.12 longitudinal shell welds resulted in an 86.7% average coverage. Two of the three upper shell longitudinal received 62% and 80% examinations; limitations were due to intersecting nozzles. The three middle shell longitudinal welds received 100% examinations. The three lower shell longitudinal welds each received an 80% examination; limitations were due to core support lugs.

Another consideration, is that, as stated by the licensee, 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.

Considering the extent to which the examinations were performed on these welds and the leakage monitoring and tests, it is reasonable to conclude that the examination coverage was sufficient to detect any existing patterns of degradation. The results from the examination of a significant portion of the weld volume indicate that the vessel is not experiencing any safety-significant service-related degradation.

3.0 CONCLUSION

Based on the information submitted, the INEL staff concludes that, pursuant to 10 CFR 50.55a(g)(6)(ii)(A), the licensee's proposed alternative to the augmented RPV examination requirements, i.e., examination of the of the accessible volume from the ID surface, provides an acceptable level of quality and safety.