UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE TO 10 CFR 50.55A(g)(6)(ii)(A)

AUGMENTED REACTOR PRESSURE VESSEL EXAMINATION

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NUMBER: 50-316

1.0 INTRODUCTION

The Technical Specifications for the Donald C. Cook Nuclear Plant, Unit 2 state that the inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2 and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of the ASME (B&PV) Code Section XI during the Donald C. Cook Nuclear Plant, Unit-2 second 10-year (ISI) interval was the 1983 Edition through the Summer 1983 Addenda. The applicable edition of Section XI of the ASME (B&PV) Code for the Donald C. Cook Nuclear Plant, Unit-2 third 10-year ISI interval is the 1989 Edition. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME (B&PV) Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME (B&PV) Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement.

After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed.

In accordance with 10 CFR 50.55a(g)(6)(ii)(A), all licensees must implement once, as part of the inservice inspection interval in effect on September 8, 1992, an augmented volumetric examination of the RPV welds specified in Item B1.10 of Examination Category B-A of the 1989 Edition of the ASME (B&PV) Code, Section XI. Examination Category B-A, Items B1.11 and B1.12 require volumetric examination of essentially 100% of the RPV circumferential and longitudinal shell welds, as defined by Figures IWB-2500-1 and -2, respectively. Essentially 100%, as defined by 10 CFR 50.55a(g)(6)(ii)(A)(2), is greater than 90% of the examination volume of each weld. Pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), licensees that are unable to completely satisfy the requirements for augmented reactor vessel shell weld examination shall propose an alternative that would provide an acceptable level of quality and safety. That alternative may be authorized by the NRC. By letter dated July 29, 1996, Indiana Michigan Power Company submitted its alternative to 10 CFR 50.55a(g)(6)(ii)(A), augmented reactor pressure vessel examination requirements for the Donald C. Cook Nuclear Plant, Unit 2.

2.0 EVALUATION

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the information provided by the licensee in support of its alternative to 10 CFR 50.55a(g)(6)(ii)(A), augmented reactor pressure vessel examination requirements for the Donald C. Cook Nuclear Plant, Unit 2.

Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the Technical Letter Report attached. Conclusions drawn from the Technical Letter Report are applicable for both the 1983 Edition through the Summer 1983 Addenda and the 1989 Edition of the ASME (B&PV) Code, Section XI. At Donald C. Cook Nuclear Plant, Unit 2, the augmented coverage requirements cannot be met for the lower shell-to-head weld and three longitudinal shell welds. The lower head-to-shell weld was examined to the extent possible, obtaining 80% volumetric coverage. Anti-rotation lugs limited scanning and thereby precluded the required coverage. For Welds RPV-VA1, RPV-VA2, and RPV-VA3, volumetric coverage was limited to 90% due to interferences from the inlet and outlet nozzles. To achieve complete volumetric coverage for the subject welds, design modifications of the component would be required. Additionally, past examinations of the accessible RPV shell welds have revealed no recordable indications.

The licensee obtained extensive coverage for each weld with a cumulative coverage of 94%. These examinations revealed no recordable flaws, and the staff concluded that inservice degradation, if present, would have been detected. The reduction in the required examination coverage is acceptable because the reactor coolant system is designed and constructed to have a low probability of gross rupture or significant leakage throughout its design life. The most likely weld failure would be a crack that would allow 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.

The licensee 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 was predicted. The examination from the RPV outside surface would cause significant undue hardship and burden without compensating increase in safety.

Based on the above evaluation, the staff determined that the examination of the accessible weld volume provides sufficient and reasonable assurance of vessel integrity. The reduction in the required 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. Therefore, the staff concludes that the licensee's proposed alternative, in combination with future examinations required by Code, provides an acceptable level of quality and safety.

3.0 CONCLUSIONS

The staff has concluded that the licensee is unable to meet the coverage requirements of 10 CFR 50.55a(g)(6)(ii)(A) for certain welds. This is a hardship without a compensating increase in safety since the proposed alternative, the augmented reactor pressure vessel examination of the accessible weld volumes from the inside surface in combination with future reactor pressure vessel examinations required by the Code, provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) and 10 CFR 50.55a(g)(6)(ii)(A)(5).

Attachment: Technical Letter Report

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Date: October 7, 1998

TECHNICAL LETTER REPORT
ALTERNATIVE TO 10 CFR 50.55a(g)(6)(ii)(A)
AUGMENTED REACTOR PRESSURE VESSEL EXAMINATION
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT UNIT 2
DOCKET NUMBER 50-316

1.0 INTRODUCTION

By letter dated July 29, 1996, the licensee, Indiana Michigan Power Company, proposed an alternative to the augmented examination of the reactor pressure vessel (RPV) required by 10 CFR 50.55a(g)(6)(ii)(A) for Donald C. Cook Nuclear Plant, Unit 2. The Idaho National Engineering Laboratory (INEL) staff has evaluated the information provided by the licensee regarding this alternative in the following section.

2.0 EVALUATION

The licensee performed the augmented reactor pressure vessel weld examinations during the 1996 refueling outage. The information provided by the licensee in support of the proposed alternative has been evaluated and the basis for disposition is documented below.

Alternative to 10 CFR 50.55a(g)(6)(ii)(A), Augmented Reactor Pressure Vessel Examination

Regulatory Requirement: In accordance with 10 CFR 50.55a(g)(6)(ii)(A), all licensees must implement once, as part of the inservice inspection interval in effect on September 8, 1992, an augmented volumetric examination of the RPV welds specified in Item B1.10 of Examination Category B-A of the 1989 Edition of the ASME Code, Section XI. Examination Category B-A, Items B1.11 and B1.12 require volumetric examination of essentially 100% of the RPV circumferential and longitudinal shell welds, as defined by Figures IWB-2500-1 and -2, respectively. Essentially 100%, as defined by 10 CFR 50.55a(g)(6)(ii)(A)(2), is greater than 90% of the examination volume of each weld.

Licensee's Proposed Alternative: The licensee proposes that the coverages obtained for the subject welds be found acceptable. The licensee obtained 100% examination coverage of each RPV shell weld, as required by 10 CFR 50.55a(g)(6)(ii)(A), with the exception of

Welds RPV-D, RPV-VA1, RPV-VA2, and RPV-VA3. The cumulative coverage for the welds subject to the augmented requirements is 94%. The coverages for all the welds are listed below.

Circumferential Item B1.11		Longitudinal Item B1.12	
Weld ID	% Examined	Weld ID	% Examined
RPV-B	100	RPV-VA1	90
RPV-C	100	RPV-VA2	90
RPV-D	80	RPV-VA3	90
		RPV-VB1	100
		RPV-VB2	100
		RPV-VC1	100
		RPV-VC2	100

<u>Licensee's Basis for the Proposed Alternative</u> (as stated):

"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 nozzle integral extensions (See Table 1)[1].

"The automated RPV examination 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

[1]Tables provided by the licensee are not included with this evaluation.

coverage of these welds."

Justification for Using Alternate Examinations

"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."

Licensee's Proposed Alternative Examination (as stated):

"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."

Evaluation: To comply with the augmented reactor vessel examination requirements of 10 CFR 50.55a(g)(6)(ii)(A), licensees must volumetrically examine essentially 100% of each of the Item B1.10 shell welds. In accordance with the regulations, essentially 100% is defined as greater than 90% of the examination volume of each weld. As an alternative to the regulations, the licensee proposes that the examinations performed to the extent practical, namely, 100% coverage on all but four of the subject examination areas, be acceptable in lieu of examining at essentially 100% of each weld.

At Donald C. Cook Nuclear Plant, Unit 2, the augmented coverage requirements cannot be met for the lower shell-to-head weld and three longitudinal shell welds. The lower head-to-shell weld was examined to the extent possible, obtaining 80% volumetric coverage. Anti-rotation lugs limited scanning and thereby precluded the required coverage. For Welds RPV-VA1, RPV-VA2, and RPV-VA3, volumetric coverage was limited to 90% due to interferences from the inlet and outlet nozzles. To achieve complete volumetric coverage for the subject welds, design modifications of the component would be required.

Considering that 1) extensive coverage was obtained for each weld, 2) the cumulative coverage for the subject welds was 94%, and 3) these examinations revealed no recordable flaws, it is reasonable to conclude that inservice degradation, if present, would have been detected.

Based on the above evaluation, it is concluded that the examination of the accessible weld volume provides sufficient and reasonable assurance of vessel integrity. The reduction in the required 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. The most likely weld failure would be a crack that would allow 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. Therefore, it is concluded that the licensee's proposed alternative, in combination with future examinations required by Code, provides an acceptable level of quality and safety.

3.0 CONCLUSION

The INEL staff has reviewed the licensee's submittal on the proposed alternative to the augmented examination of the reactor pressure vessel (RPV) required by 10 CFR 50.55a(g)(6)(ii)(A) for Donald C. Cook Nuclear Plant, Unit 2. The licensee has examined a significant portion of the reactor vessel welds (94%), detecting no recordable service-related flaws. Therefore, it is concluded that the licensee's proposed alternative, in combination with future reactor pressure vessel examinations, provides an acceptable level of quality and safety. It is further concluded that additional examinations from the OD are not possible due to inaccessibility caused by the bioshield wall. Therefore, it is recommended that the licensee's proposed alternative be authorized in accordance with 10 CFR 50.55a(g)(6)(ii)(A)(5).