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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF REGARDING RPV BELTLINE WELDS

FOR

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NUMBER: 50-333

1.0 INTRODUCTION

The Technical Specifications for James A. FitzPatrick Nuclear Power Plant, state that the inservice inspection and testing 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 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). The Code of Federal Regulations of 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 difficulties 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 preservice examination requirements, set forth in the ASME 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) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable Edition of Section XI of the ASME Code for the James A. FitzPatrick Nuclear Power Plant second 10-year inservice inspection (ISI) interval is the 1980 Edition, through Winter 1981 Addenda. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME 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 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.

On January 31, 1991, the NRC published, for comment, a proposed rule for an augmented examination of reactor pressure vessel welds. The proposed rule would have revoked previously granted relief requests for reactor pressure vessel weld examinations. However, the final rule, effective September 8, 1992, permits licensees with fewer than 40 months remaining in the 10-year inspection interval in effect on September 8, 1992, to defer the augmented reactor vessel weld examinations to the first period of the next inspection interval. Further, the final rule permits licensees that defer the augmented examination to retain all previously granted reliefs for the inspection interval in effect on September 8, 1992. Since the FitzPatrick plant was in the last period when the rule was implemented, the licensee deferred augmented examinations until the first period of the next inspection interval, as permitted by the regulations. These augmented examinations will include essentially 100% of all reactor vessel shell welds. In addition, as a result of the final rule, the licensee in a letter dated July 30, 1993, resubmitted an alternative examination regarding examination of the reactor pressure vessel beltline welds for the second 10-year ISI interval, which ends July 28, 1995.

2.0 EVALUATION AND CONCLUSIONS

The NRC 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 examination contained in its request for relief regarding inservice examination of the reactor pressure vessel beltline welds for the second 10-year ISI interval. Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the Technical Evaluation Summary attached. The staff concludes that the proposed alternative for ISI examination of the reactor pressure vessel beltline welds will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative may be authorized on a one-time basis only, provided that the new regulations are met during the first inspection period of the third 10-year interval.

Principal Contributor: T. McLellan

Date: March 8, 1994

ENCLOSURE 2

TECHNICAL EVALUATION SUMMARY OF THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION REQUEST FOR RELIEF - RPV BELTLINE WELDS FOR POWER AUTHORITY OF THE STATE OF NEW YORK JAMES A. FITZPATRICK NUCLEAR POWER PLANT DOCKET NUMBER: 50-333

1.0 INTRODUCTION

In a letter dated July 30, 1993, the licensee, the Power Authority of the State of New York (PASNY), submitted in its request for relief a proposed alternative examination to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements regarding inservice examination of the reactor pressure vessel beltline welds for the second 10-year inservice inspection (ISI) interval, which ends July 28, 1995. The Idaho National Engineering Laboratory (INEL) staff has evaluated the subject request in the following section.

2.0 EVALUATION

The information provided by the licensee in support of the alternative to the Code requirements has been evaluated and the bases for authorizing the alternative are documented below. The applicable edition of Section XI of the ASME Code for the James A. FitzPatrick Nuclear Plant, second 10-year inservice inspection (ISI) interval, is the 1980 Edition, through Winter 1981 Addenda.

A. Relief Request for Inservice Inspection of the Reactor Pressure Vessel Beltline Welds, Examination Category B-A, Items B1.11 and B1.12

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Items B1.11 (circumferential shell welds) and B1.12 (longitudinal shell welds) require a 100% volumetric examination, as defined by Figures IWB-2500-1 and IWB-2500-2 respectively, of one circumferential and one longitudinal beltline region weld.

Licensee's Code Relief Request: The licensee has requested relief from the Code-requirement to examine 100% of the length of one beltline region circumferential weld and 100% of the length of one beltline region longitudinal weld. The licensee stated that this relief would apply only to the current inspection interval.

Licensee's Basis for Requesting Relief: The licensee stated: "The design of the biological shield wall and permanent insulation system does not allow access from the outside of the reactor vessel to 100% of one longitudinal weld and 100% of one circumferential weld. As shown on drawing MSK-3036 (enclosed[1]) the removable insulation panels permit access to only a portion of the vessel shell welds. The biological shield wall and insulation would have to be redesigned and refabricated, at an enormous cost in both dollars and personnel radiation exposure, to complete the volumetric examinations as required by the Code from the outside of the vessel. Accessibility restraints was the basis for NRC approved relief for many other plants which limit their inspection of welds accessible from the outside of the vessel.

There are currently only a limited number of systems designed to inspect welds from inside the vessel. This creates uncertainty regarding their availability for use during the next FitzPatrick refueling outage scheduled for January 1995, considering the expected high demand from utilities required to perform the augmented inspection during their current inspection interval. Use of the in-vessel inspection system would create an economic hardship by adding 7 days of critical path time to the length of the outage.

The alternative examinations proposed would subject a composite of longitudinal beltline and non-beltline region welds, that are equivalent in length to one longitudinal beltline weld (approximately 150 inches), to examination. The longitudinal beltline welds accessible for inspection (approximately 60 inches) are located in three equally spaced welds, and represent approximately 40% of the longitudinal welds subject to the alternative examination. The examinations are, therefore, representative of beltline region conditions, and meet the intent of the Code.

Likewise, the alternative examinations proposed would subject a composite of circumferential beltline welds, and non-beltline region circumferential and longitudinal welds, that are equivalent in length to one circumferential beltline weld (approximately 730 inches), to an examination. The circumferential beltline welds accessible for inspection (approximately 111 inches) are located in three equally spaced segments, and represent approximately 15% of the welds subject to the alternative examination. The examinations are, therefore, representative of beltline region conditions, and meet the intent of the Code.

To ensure reactor vessel stresses remain within acceptable limits, the plant is subject to pressure/temperature operating limits for reactor coolant system heatup and cooldown operations. The pressure/temperature operating limits are based on the evaluation of vessel materials for radiation damage performed in accordance with the more stringent

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[1] Drawings referenced in licensee's basis are not included in this document.

requirements recently imposed by Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials." Further, the augmented examination of vessel welds will be performed during the next inspection interval in accordance with a recent regulation [10 CFR 50.55a(g)(6)(ii)(A)], complementing the alternative examinations proposed for the current inspection interval.

Inservice inspection previously performed on FitzPatrick Reactor Pressure Vessel welds have shown the vessel weld materials to be free of cracks. During the 1988, 1990, and 1992 refueling outages, 2558 inches of RPV welds were examined (57 inches in the beltline region)."

Licensee's Proposed Alternative Examination: The licensee proposed that circumferential weld (No. VC-3-4) in the beltline region will be examined to the extent that the weld is accessible from the outside surface of the vessel (approx. 111 inches). Additionally, non-beltline circumferential and non-beltline longitudinal welds, accessible from the outside surface of the vessel, will be examined so that the total composite length examined during the current 10-year inspection interval is equivalent to the length of one beltline region circumferential weld (730 inches).

Longitudinal welds in the beltline region (VV-3A, VV-3B, and VV-3C) will be examined to the extent that they are accessible from the outside surface of the vessel (approx. 60 inches). Additionally, non-beltline longitudinal welds, accessible from the outside surface of the vessel, will be examined so that the total composite length examined during the current 10-year inspection interval is equivalent to the length of one beltline region longitudinal weld (150 inches).

Staff Evaluation: The subject relief request was incorporated in the second 10-year inspection interval Inservice Inspection Program submitted September 30, 1985, and supplemented by letter dated December 8, 1986. The staff approved the program plan except for several reactor vessel weld relief requests, including the welds contained in this relief request. At that time, the staff denied approval on the basis that all reactor pressure vessel requests for relief were under staff review anticipating a change in the regulations. Previous to that time, the staff had granted similar relief for other plants.

On January 31, 1991, the NRC published, for comment, a proposed rule for an augmented examination of reactor pressure vessel welds. The proposed rule would have revoked previously granted relief requests for reactor pressure vessel weld examinations. However, the final rule, effective September 8, 1992, permits licensees with fewer than 40 months remaining in the 10-year inspection interval in effect on September 8, 1992, to defer the augmented reactor vessel weld examinations to the first period of the next inspection interval. Further, the final rule permits licensees that defer the augmented examination to retain all previously granted reliefs for the inspection interval in effect on September 8,

1992. Since the FitzPatrick plant was in the last period when the rule was implemented, the licensee deferred augmented examinations until the first period of the next inspection interval, as permitted by the regulations. These augmented examinations will include essentially 100% of all reactor vessel shell welds. As a result of the final rule, the licensee resubmitted the request for relief regarding the examination of the beltline welds.

The design of the biological shield wall and permanent insulation does not allow access from the outside of the reactor vessel to perform the Code-required 100% volumetric examination of one circumferential and one longitudinal beltline weld. The new regulations require FitzPatrick to perform augmented 100% volumetric examinations on all reactor vessel shell welds during the first period of the third 10-year interval. The new regulations will necessitate that the examinations be performed from the inside of the vessel and, the licensee expects the examinations are to be in full compliance with the Code.

The alternative examinations proposed by the licensee, for the second 10-year interval, would consist of a composite of beltline and non-beltline welds such that the resulting length would be equal to the Code-required length of a single circumferential (approx. 730 inches) and a single longitudinal (approx. 150 inches) beltline weld. These examinations would be representative of the beltline region conditions. Thus, reasonable assurance of the structural integrity of the subject reactor pressure vessel welds will be provided for the interim period until the augmented examination is performed.

3.0 CONCLUSION

We have reviewed the licensee's submittal and have concluded that the proposed alternative for ISI examination of the reactor pressure vessel beltline welds will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternatives should be authorized on a one-time basis only, provided that the new regulations are met during the first inspection period of the third 10-year interval.

10 CFR 50.55a(a)(3)(i), PASNY's proposed alternative is authorized on a one-time basis only, provided that the new regulations, as stated in the enclosed SE, are met during the first inspection period of the third 10-year interval.

This completes our action related to TAC No. M87158.

Sincerely,

Original signed by:

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation
2. Technical Evaluation Summary

DISTRIBUTION:

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PDI-1 Reading	LReyes, 12/G/18
TMurley/FMiraglia, 12/G/18	ERossi, 9/A/2
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JCalvo	RACapra
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