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

ALTERNATIVE TO THE AUGMENTED EXAMINATION OF THE REACTOR VESSEL

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

On May 28, 1998, as supplemented on September 10, 1998, and revised on September 29, 1998, the Power Authority of the State of New York, (the licensee, also known as the New York Power Authority) requested an alternative to performing the reactor pressure vessel (RPV) shell weld examination requirements of both the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 1989 Edition, and the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2), for the James A. FitzPatrick Nuclear Power Plant. Licensees are required by 10 CFR 50.55a(g)(6)(ii)(A) to perform an expanded RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code, on an "expedited" basis. The licensee's propose alternative was submitted pursuant to the provisions of 10 CFR 50.55a(g)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i).

Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class 1, 2 and 3 components must 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. This regulation requires 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 the ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The licensee's May 28, 1998, submittal requested approval of an alternative plan for RPV shell weld examinations planned for the 1998 refueling outage 13. Supplemental information regarding these plans was provided on September 10, 1998. The planned inspections fell well short of the goal given in 10 CFR 50.55a(g)(6)(ii)(A)(2) to obtain more than 90 percent of the examination volume for each weld, particularly in the vessel beltline region where neutron fluence is highest. As a result, the licensee revised its request on September 29, 1998, proposing deferral of the inspections for one operating cycle (refueling outage 14, currently planned for the fourth quarter of 2000). This deferral will allow the licensee to review and plan alternative methods that will allow inspection a much greater volume of each RPV shell weld. The licensee has made a commitment to submit a letter to the NRC regarding improvements made for performing augmented examination of the RPV vertical shell welds to the maximum extent possible, including details on examination coverage of each weld, specifically detailing vertical weld length coverage in the belt line region by December 31, 1999.

In Information Notice 97-63, Supplement 1, "Status of NRC Staff's Review of BWRVIP-05," the staff indicated that it would consider technically justified alternatives to the augmented examination in accordance with 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), and 10 CFR 50.55a(g)(6)(ii)(A)(5), for BWR licensees scheduled to perform inspections of the BWR RPV circumferential welds during the fall 1998 or spring 1999 outage seasons. Acceptably justified alternatives would be considered for inspection delays of up to 40-months or two operating cycles (whichever is longer) for BWR RPV circumferential shell welds.

2.0 BACKGROUND

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, June 13 and December 18, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners Group (BWROG), submitted the proprietary report, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," which proposed to reduce the scope of inspection of the BWR RPV welds from essentially 100 percent of all RPV shell welds to 50 percent of the vertical welds and 0 percent of the circumferential welds. On October 29, 1996, the BWRVIP modified their proposal to increase the examination of the vertical welds to 100 percent from 50 percent, while still proposing to inspect essentially 0 percent of the circumferential RPV shell welds, except that the intersection of the vertical and circumferential welds would have included approximately 2-3 percent of the circumferential welds.

On May 12, 1997, the NRC staff and members of the BWRVIP met with the Commission to discuss the staff's review of the BWRVIP-05 report. In accordance with guidance provided by the Commission in Staff Requirements Memorandum (SRM) M970512B, dated May 30, 1997, the staff initiated a broader, risk-informed review of the BWRVIP-05 proposal, and issued an interim independent staff assessment and a final safety evaluation related to the review of BWRVIP-05 on August 14, 1997 and July 28, 1998, respectively. The final safety evaluation concluded that elimination of examination of circumferential RPV shell welds was justified. Note that this final safety evaluation was issued after the licensee's May 28, 1998, submittal of a proposed alternative examination.

The staff's Independent Assessment of the BWRVIP-05 proposal is documented in a letter dated August 14, 1997, to Mr. Carl Terry, BWRVIP Chairman. The staff concluded that the industry's assessment did not sufficiently address risk, and additional work was necessary to provide a complete risk-informed evaluation. The staff's assessment was performed for BWR RPVs fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The staff assessment identified pressure and temperature resulting from a cold overpressure event in a foreign reactor as the limiting event that could cause BWR RPV failure. The materials and neutron radiation parameters used by the staff are identified in Table 7-1 of the staff's independent assessment. The staff determined the conditional probability of failure for vertical and circumferential welds fabricated by CB&I, CE and B&W. Table 7-9 of the staff's assessment identifies the conditional probability of failure for the reference cases and the 95% confidence uncertainty bound cases for vertical and circumferential welds fabricated by CB&I, CE and B&W. B&W fabricated vessels were determined to have the highest conditional probability of failure. The input material parameters used in the analysis of the reference case for B&W fabricated vessels resulted in a reference temperature for nil ductile transition (RT_{NDT}) at the vessel inner surface of 114.5 °F. In the uncertainty analysis, the neutron fluence evaluation had the greatest RT_{NDT} value (145 °F) at the inner surface. Vessels with RT_{NDT} values less than those resulting from the staff's assessment will have less embrittlement than the vessels simulated in the staff's assessment and should have a conditional probability of vessel failure less than or equal to the values in the staff's assessment.

The failure probability for a weld is the product of the critical event frequency and the conditional probability of the weld failure for that event. The FitzPatrick RPV was fabricated by Combustion Engineering. Using the event frequency for a cold overpressure event and the conditional probability of vessel failure for CE fabricated circumferential welds, the best-estimate failure frequency from the staff's assessment is <6x10 $^{-11}$ [(1)] per reactor year and the upper bound failure frequency from the uncertainty analysis is <2.8x10 $^{-10}$ [(1)] per reactor year.

3.0 PROPOSED ALTERNATIVE

The alternative proposed by the licensee on September 29, 1998 is as follows:

> The alternative plan would defer the augmented exams to refueling outage 14 (currently scheduled for 4th quarter 2000). The [licensee] will evaluate methods for performing RPV vertical weld examinations to the maximum extent possible and provide greater than 90 percent coverage of the vertical welds in the belt-line region, and incidental coverage of 2-3 percent of the intersecting circumferential welds. Further examination of the circumferential welds would depend on NRC review, resolution, and approval of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) recommendations contained in "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," BWRVIP-05. The proposed deferral is an alternative to the augmented examinations for RPV shell welds specified in 10 CFR 50.55a(g)(6)(ii)(A)(2).

4.0 LICENSEE'S TECHNICAL JUSTIFICATION

NYPA is unable to meet the greater than 90 percent coverage required for each weld due to interference of the FitzPatrick RPV internals and other components and the examination equipment's lower scan limitations. The alternative proposed in the letter of May 28, 1998, was perform an examination of the RPV shell welds to the maximum extent practical from the inner diameter (ID), within the constraints of vessel internal interferences. Accessibility studies of the FitzPatrick RPV have determined that the accessible area for volumetric examinations from the ID will allow coverage of approximately 60 percent of the cumulative length of the shell welds (vertical and circumferential welds). This would have only allowed coverage of approximately 33 percent of the cumulative length of the vertical welds in the belt line region. Further examination from the ID is not possible without disassembly of vessel internals and other components. In order to develop methods capable of more comprehensive weld inspections, on September 29, 1998, the licensee proposed to defer the RPV weld inspections for a single operating cycle.

During the fabrication process of the RV, all of the shell welds were thoroughly examined using several examination methods as required by the original construction code. Additionally, all of the shell welds received volumetric examinations prior to initial plant operations, as prescribed by ASME Section XI pre-service inspection requirements.

[(1)] Insufficient or no failures to accurately determine reference case failure probability and sensitivity to flaw size, flaw density and inservice inspection.

Selected shell welds have received outer diameter (OD) volumetric examinations during the first and second inservice inspection interval in accordance with ASME Section XI inservice inspection requirements. Only minor non-crack indications were identified during the first and second inservice interval examinations. The indications were found acceptable for operation.

General Electric provided the results of previous examinations performed on BWRs. According to the licensee, the results show that significant indications are not prevalent in the RPV shell welds for the industry as a whole and those found were determined to be acceptable for operation.

5.0 EVALUATION

During review of the BWRVIP-05 report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," the staff identified non-design basis events which should have been considered in the BWRVIP-05 report. In particular, the potential for and consequences of cold overpressure transients should be considered. The licensee assessed the systems that could lead to a cold overpressurization of the FitzPatrick RPV. These included the high pressure core injection (HPCI), reactor core isolation cooling (RCIC), standby liquid control (SLC), feedwater and feedwater booster pumps, low pressure core spray (LPCS), low pressure core injection (LPCI), control rod drive (CRD) and reactor water cleanup systems (RWCU).

The HPCI, RCIC, and feedwater pumps are steam driven and do not function during cold shutdown. No steam is available for operation of these systems during cold shutdown. The licensee stated that there were no automatic starts associated with SLC. Operator initiation of SLC should not occur during shutdown, however, the SLC injection rate is approximately 50 gpm which would allow operators sufficient time to control reactor pressure if manual initiation occurred.

The feedwater booster pumps are maintained in the pulled to lock position during cold shutdown. In this condition, several operator errors and breakdowns in work control processes would have to occur to inadvertently start a feedwater booster pump and inject into the vessel. The LPCS and LPCI systems are low pressure ECCS systems with low shutoff heads. If either one of these systems were manually or inadvertently initiated during cold shutdown, the resulting reactor pressure and temperature would be below the pressure-temperature limits. The CRD and RWCU systems use a feed and bleed process to control RPV level and pressure during normal cold shutdown conditions. Per plant procedures, the reactor head vents are open when reactor coolant temperature is less than 212 °F. The CRD pumps injection rate is less than 60 gpm; this flowrate and the opened reactor head vents allow sufficient time for operators to react to unanticipated level changes.

In all cases, the operators are trained in methods of controlling water level within specified limits in addition to responding to abnormal water level conditions during shutdown. The licensee also stated that procedural controls for reactor temperature, level and pressure are an integral part of operator training. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification pressure-temperature limits. On the basis of the pressure limits of the operating systems, operator training, and established plant-specific procedures, the licensee determined that a non-design basis cold over-pressure transient is unlikely to occur during the requested delay.

In the letter dated August 14, 1998, to Carl Terry, BWRVIP Chairman, transmitting the staff's Independent Assessment of the BWRVIP-05 report, it is stated that the acceptability of requests for delays in inspections will be based on plant-specific information submitted by the licensee that demonstrates that the expected frequency of beyond design basis events and the level of embrittlement are low enough to assure low probability of vessel failure during the period of relief.

The licensee provided the following information to show the conservatism of the NRC analysis for FitzPatrick at an estimated 19 effective fuel power years (EFPY):

> For plants with RPVs fabricated by Combustion Engineering (CE), the mean end-of-license neutron fluence (32 EFPY) used in the NRC analysis...was 0.15E+19 n/cm^2. However the highest fluence anticipated for FitzPatrick NPP [Nuclear Power Plant] at the end of the next two operating cycles (19 EFPY) is 9.56E+17 n/cm^2. The projected fluence for the FitzPatrick plant for 19 EFPY (October 2002, an additional 2 years past the requested deferral period) is considerably less, with regard to the effect of fluence on embrittlement, than the NRC analysis.

The licensee's discussion of neutron fluences does not show that the level of embrittlement for the FitzPatrick RPV are less than the values in the BWRVIP-05 report or the staff's Independent Assessment since embrittlement is a function of both neutron fluence and weld chemistry (i.e., copper and nickel). Therefore, to determine the level of embrittlement of the FitzPatrick RPV circumferential shell welds the staff calculated the embrittlement projected for the beltline circumferential shell weld at 19 EFPY (the table below summarizes the staff's calculations of the level of embrittlement).

PARAMETER DESCRIPTION	FitzPatrick COMPARATIVE PARAMETERS AT 19 EFPY (Bounding Circ. Weld)	NRC INDEPENDENT ASSESSMENT LIMITING FRACTURE ANALYSIS PARAMETERS
Fluence, n/cm^2	9.56×10^{17}	3.26×10^{18}
Initial RT_{NDT}, °F	-50	-56
Chemistry Factor	209.1	177.6
Cu%	0.337	0.226
Ni%	0.609	0.76
ΔRT_{NDT}	85.3	122.9
Margin Term	42.65	61.5
Mean Adjusted Reference Temperature (ART)	35.3	66.9
Upper Bound ART	77.95	128.4

The NRC staff has conducted an independent risk-informed assessment of the analysis contained in BWRVIP-05. This independent NRC assessment utilized the FAVOR Code to perform a probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. The key parameters in the PFM analysis are the initial $RT_{NDT,}$ the end of license mean neutron fluence, the mean chemistry (percent copper and nickel) of the welds, and the pressure and temperature of the events being considered. Although according to the licensee, BWRVIP-05 provides the technical basis supporting their alternative, the previous table, which was developed by the staff, illustrates that the FitzPatrick RPV has additional conservatism in comparison to the NRC's Independent Assessment Fracture Analysis limiting case. The staff's analysis was based on the staff's Independent Assessment of BWRVIP-05.

The chemistry factor, ΔRT_{NDT}, margin term, mean ART, and upper bound ART are calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2. Since the upper bound ART for the bounding FitzPatrick circumferential weld is less than the value from the NRC Independent Assessment, the staff confirmed that the FitzPatrick circumferential weld is bounded by the staff's assessment, thus providing additional assurance that the vessel welds are also bounded by the BWRVIP-05 report.

The staff calculated that the RT_{NDT} values for the circumferential welds at the end of the relief period are less than the values in the reference case and uncertainty analysis for the CE fabricated vessels. RT_{NDT} is a measure of the amount of irradiation embrittlement. Since the RT_{NDT} values are less than the values in the reference case and the uncertainty analysis for CE fabricated vessels, the FitzPatrick RPV will have less embrittlement than the reference case and will have a conditional probability of vessel failure less than or equal to that estimated in the staff's assessment.

The staff reviewed the information provided by the licensee regarding the FitzPatrick high pressure injection systems, operator training, and plant-specific procedures to prevent RPV cold over-pressurization. The information provided sufficient basis to support approval of the alternative examination request. The staff concludes that the probability of a non-design basis cold over-pressure transient occurring at FitzPatrick during the requested delay is low, which is consistent with the staff's assessment.

The amount of increase in embrittlement for a given period of time of irradiation is determined upon the increase in the reference temperature, RT_{NDT}. The increase in reference temperature is a function of both neutron fluence and weld chemistry. In the case of FitzPatrick, the vertical welds were fabricated with two different heats of weld wire. The best estimate chemistry for heat 27204/12008 is 0.219% Cu and 0.996% Ni and for weld heat 13253/1202 it is 0.210% Cu and 0.873% Ni. The staff extrapolated the fluence, based on the licensee's value for 19 EFPY, and estimated the increase in reference temperature for the 1 year-extension period. The staff determined that the reference temperature for the vertical weld would only increase by less than 5 °F. Since this is a very small increase in the RT_{NDT}, the RPV vertical shell welds will have adequate margin to brittle fracture during the additional operating cycle.

6.0 CONCLUSION

Based upon its review, the staff reached the following conclusions:

1. Based on the licensee's and staff's assessment of the materials in the circumferential welds in the FitzPatrick RPV, the conditional probability of vessel failure should be less than or equal to that estimated from the staff's and BWRVIP-05 assessments.

2. Based on the licensee's high pressure injection systems analyses, operator training, and plant-specific procedures, the probability of a non-design basis cold overpressure transient is low during the requested delay period and is consistent with the staff's assessment.

3. Based on the small amount of additional embrittlement that will occur in the FitzPatrick PV vertical shell welds during the next operating cycle, it is concluded that the FitzPatrick RPV vertical shell welds will have adequate margin to brittle fracture during the next operating cycle.

4. Based on the above three conclusions, the staff concludes that the FitzPatrick RPV can be operated during the requested delay period with an acceptable level of quality and safety, and the inspection of the circumferential and vertical welds may be delayed for the requested one operating cycle.

Therefore, the proposed postponement of beginning the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2) at FitzPatrick for circumferential and vertical shell welds for one operating cycle is authorized pursuant to 10 CFR 50.55a(a)(3)(i), because the alternative to the ASME Code requirements provides an acceptable level of quality and safety.

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