James A. FitzPatrick
Nuclear Power Plant
268 Lake Road
P.O. Box 41
Lycoming, New York 13093

315-342-3840

New York Power Authority

Michael J. Colomb
Site Executive Officer

September 29, 1998
JAFP-98-0316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: James A FitzPatrick Nuclear Power Plant
Docket No. 50-333
Proposed Alternatives in Accordance with 10CFR50.55a(a)(3)(i) for Reactor Pressure Vessel Shell Weld Examinations

Reference: NYPA letter, J Knubel, to NRC, "Proposed Alternatives in Accordance with 10CFR50.55a(a)(3)(i) for Reactor Pressure Vessel Shell Weld Examinations," (JPN-98-22) dated May 28, 1998.

Dear Sir:

This letter transmits a request for NRC approval of an alternative plan, in accordance with 10CFR50.55a(a)(3)(i), for the Reactor Pressure Vessel (RPV) shell weld examinations, pursuant to the provisions of 10CFR50.55a(g)(6)(ii)(A)(5). This letter revises a previous submittal (Reference) that requested a deferral of the requirement to perform circumferential RPV shell weld inspections for two cycles and contained an alternative plan for performing the vertical RPV shell weld exams during the upcoming refuel outage (R13). The proposed alternative for the RPV shell weld augmented examination specified in 10CFR50.55a(g)(6)(ii)(A)(2), is to defer the examination one operating cycle based on the attached basis that shows an acceptable level of quality and safety will be provided. NRC approval of this alternative will allow the Authority to review and plan alternative methods that will allow greater access and inspection of the specified welds to satisfy the requirements to inspect essentially 100 percent of the examination volume of each RPV shell weld.

Attachment 1 contains the Authority's supporting justification and basis for the alternative plan for the RPV shell weld examinations for the FitzPatrick plant. Review and approval of this alternative plan is requested prior to October 15, 1998.

A047/1

If you have any questions please contact Mr. Art Zaremba at (315) 349-6365.

Very truly yours,

acting

MICHAEL J. COLOMB
Site Executive Officer

cc: Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
James A. FitzPatrick Nuclear Power Plant
U.S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, NY 130931

Mr. Joseph Williams, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
U.S. Nuclear Regulatory Commission
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Washington, DC

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Background:

10CFR 50.55a(g)(6)(ii)(A)(2) states that all licensees shall augment their reactor vessel examinations by implementing the examination requirements for Reactor Pressure Vessel (RPV) shell welds specified in item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in 50.55a(g)(6)(ii)(A)(3) and (4). As stated in 10CFR50.55a(g)(6)(ii)(A)(2) for the purposes of this augmented examination, essentially 100 percent as used in Table IWB-2500-1 means more than 90 percent of the examination volume for each weld. Additionally, 10CFR50.55a(g)(6)(ii)(A)(5) requires licensees that are unable to completely satisfy the augmented RPV shell weld examination requirement to submit information to the U.S. Nuclear Regulatory Commission to support the determination, and propose an alternative to the examination requirements that would provide an acceptable level of quality and safety. The Authority is unable to obtain essentially 100 percent of each weld without disassembly or removal of internal interference, removal of permanently installed bio-shield, or modification of the inspection equipment. Accessibility studies indicate that the RPV shell weld examination coverage utilizing the GERIS 2000 equipment without RPV internal interference removal is a total of approximately 51 percent for the vertical welds, and only 33 percent of the vertical weld length in the belt-line region. The Authority's intention is to review and evaluate methods to allow accessibility to greater than 90 percent of the vertical RPV shell welds in the belt-line region. The alternative plan would allow time for review and evaluation of alternatives that could provide greater vertical weld examination coverage and ensure an acceptable level of safety and quality. The alternative plan, however, would exceed the time provisions, for completion of the augmented exams, specified in 50.55a(g)(6)(ii)(A)(2) and (3).

The purpose of this letter is to request approval, pursuant to provisions contained in 10CFR50.55a(a)(3)(i), of an alternative plan for performing the reactor pressure vessel (RPV) augmented examination requirements of 10CFR55a(g)(ii)(A)(2) for the James A. FitzPatrick Nuclear Power Plant. The Authority's alternative plan would defer the augmented exams to refueling outage 14 (currently scheduled for 4th quarter 2000). The Authority 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 (Reference 1). This is consistent with NRC Information Notice 97-63, Supplement 1. The Authority endorses the current BWRVIP recommendations contained in BWRVIP-05. If the recommendations of the BWRVIP are changed during the approval process, the Authority will reevaluate the planned scope of examinations described in this attachment in relation to conformance with the approved guideline.

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A. COMPONENT IDENTIFICATION:

ISI Class 1, Code Category B-A, "Pressure Retaining Welds in Reactor Vessel", item B1.10, "Shell Welds".

B. EXAMINATION REQUIREMENTS:

10CFR 50.55a(g)(6)(ii)(A)(2) states that all licensees shall augment their reactor vessel examinations by implementing the examination requirements for Reactor Pressure Vessel (RPV) shell welds specified in item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in 50.55a(g)(6)(ii)(A)(3) and (4). As stated in 10CFR50.55a(g)(6)(ii)(A)(2) for the purposes of this augmented examination, essentially 100 percent as used in Table IWB-2500-1 means more than 90 percent of the examination volume for each weld. Additionally, 10CFR50.55a(g)(6)(ii)(A)(5) requires licensees that are unable to completely satisfy the augmented RPV shell weld examination requirement to submit information to the U.S. Nuclear Regulatory Commission to support the determination, and propose an alternative to the examination requirements that would provide an acceptable level of quality and safety.

C. ALTERNATIVE TO THE EXAMINATION REQUIREMENTS

The alternative plan would defer the augmented exams to refueling outage 14 (currently scheduled for 4th quarter 2000). The Authority 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).

D. BASIS FOR ALTERNATIVE PLAN:

The Authority is unable to meet the greater than 90 percent coverage requirement for each weld due to internal interference of the JAFNPP reactor vessel components and the examination equipment's lower scan limitations. The alternative proposed in Reference 2 was to perform an examination of the RPV shell welds to the maximum extent practical from the inner diameter (ID), within the constraints of vessel internal interference. Accessibility studies (Reference 3) of the JAFNPP 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

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possible without disassembly of vessel internal components. This alternative will allow for review and planning of methods to allow greater access and inspection of the specified welds.

The industry basis document, BWRVIP-05, considered several issues related to BWR RPV integrity to provide a basis for eliminating the requirement to perform circumferential welds and the performance of only 50 percent of the vertical RPV shell welds. These issues included fabrication practices, inservice inspection data, operational issues, degradation mechanics, and probabilistic fracture mechanics analysis results. As stated in the report "Results of the evaluation performed in this report clearly demonstrate the inherent safety and integrity of BWR reactor pressure vessels." The following basis uses a similar approach to justify deferral of the required examinations to RO-14.

Previous Shell Weld Examinations

During the fabrication process of the RPV, 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.

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 interval examinations. The indications were found acceptable for operation. The extent of the second interval OD weld examinations and the indications identified were transmitted to the NRC via NYPA letter (JAFP-98-0292), from Michael J. Colomb, to NRC, dated September 10, 1998. A sketch of the previous OD exam locations was also included.

Industry Results of past exams:

The following information contained in the table was provided by General Electric to show results of previous exams performed at BWRs. 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.

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Total BWR RPV Shell Welds Examined by General Electric (GENE)

RPV Weld Type	No. Welds Examined	Welds with Indication Exceeding IWB-3510	No. of Indications Exceeding IWB-3510
Circ Weld ID	31	4	15
Vert Weld ID	80	1	1
Circ Weld OD	66	0	0
Vert Weld OD	209	2	2
Total	386	7	18

The ID exams found 16 indications total, located in 5 welds. The ID indications were found in B&W constructed vessels, BWR-4 plants with 22 years of operation. The OD exams found 2 indications in 2 welds. Both indications were found in CE constructed vessels, BWR-4 plants with 19 and 25 years of operation. All indications were determined to be construction related, evaluated to IWB-3600 and accepted for operation. FitzPatrick has a Combustion Engineering (CE) constructed vessel with approximately 22 years of operation.

Neutron Fluence/Embrittlement:

As published in the August 1992 Federal Register under supplementary information regarding the final rule, the NRC position with regard to augmented examination of reactor vessel shell welds is based on an embrittlement concern (in addition to stress corrosion cracking and service induced cracking) stemming from irradiation material test results which show that certain reactor vessel materials undergo greater radiation damage than previously expected.

The BWR Vessel and Internals Project report (BWRVIP-05), dated September 1995, stated that "Embrittlement issues are addressed in 10CFR50 Appendix G through requirements associated with upper shelf energy (USE) and the reference temperature of nil-ductility transition (RT_{NDT}). In order to account for the effects of embrittlement, adjusted reference temperatures (ARTs), defined in the initial RT_{NDT} plus the irradiation shift for fluence, are determined. It is possible that ARTs may result in pressure-temperature testing criteria that are difficult to meet due to increased temperature requirements. However, due to low BWR fluence, an unacceptable ART will not be reached, even when extended life is planned." Also, the report states that "In addition to increasing RT_{NDT} the USE of low alloy steel materials decreases with neutron exposure. However, for the relatively low fluence BWR, maintaining a USE above 50 ft-lbs is not a concern. Also, Code margins required by appendix G are satisfied at USE values as low as 35 ft-lbs and thus is not a safety concern. Based on the above, it can be seen that although irradiation embrittlement of materials can be a significant concern, its effect is minimal for the relatively low fluence environment of a BWR."

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Probabilistic Fracture Mechanics (PFM) Analysis:

Although BWRVIP-05 provides a technical basis for this relief, an independent NRC risk informed assessment of the analysis contained in the BWRVIP-05 report was conducted. The independent NRC assessment used the FAVOR code to perform probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are: the neutron fluence was estimated to be end-of-license mean fluence, the chemistry values are mean values based on vessel types, and the potential for beyond design basis events is considered.

The following is a statement contained in the "Executive Summary" of the "NRC Staff Final Safety Evaluation of BWRVIP-5 Report (Reference 6). "It should be noted that the failure frequency for axial welds cited above are relatively high, but that there are known conservatisms in these estimates. For example, these analyses were based on the assumption that the flaws in axial weld with the limiting material properties and chemistry are all located at the inside surface of the BWR RPV and at the location of peak end-of-license (EOL) azimuth fluence. Since flaws are distributed throughout the weld and EOL neutron fluence will not occur for many years, the staff has concluded that the present RPV failure frequency is substantially below that reported by the BWRVIP, and independently calculated by the staff, and is not a near-term safety concern."

The following information is provided to show the conservatism of the NRC analysis for the FitzPatrick plant at an estimated 19 EFPY. Changes in RT_{NDT} may be used as one of the means for monitoring radiation embrittlement of reactor vessel materials. For plants with RPVs fabricated by Combustion Engineering (CE), the mean end-of-license neutron fluence (32 EFPY) used in the NRC analysis contained in Reference 5 was 0.15E+19 n/cm^2. However the highest fluence anticipated for FitzPatrick NPP 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.

Stress Corrosion Cracking (SCC):

As stated in BWRVIP-05, SCC has been a concern in austenitic stainless steel piping, SCC in the Vessel has been limited to high carbon stainless steel components, creviced stainless steel or nickel based alloys, areas of extreme cold work in cladding, and alloy 182 vessel attachment welds. Due to the absence of high stress fields, the low alloy steel of the BWR vessel is resistant to SCC. As stated above, SCC has been observed at vessel attachment welds, however, growth into the low alloy steel has been limited to the area of high stress such as nozzle safe ends.

For FitzPatrick, visual exams of attached welds performed over the last 2 outages (RO-11 and 12) have resulted in no significant indications. The results infer that lack of indications in the higher stress areas indicate that it is unlikely that stress corrosion cracking would have developed in the low stress area of the RPV vessel welds. Attachment 2 contains the results of visual exams performed on various attached welds over the last 2 outages.

Also, the BWRVIP-05 report states that significant service induced cracking which has occurred in large vessels designed and fabricated to the ASME Code has been limited to PWRs. No instances of significant service induced cracking of BWR pressure vessel low alloy material have been identified.

Cold Over-Pressurization:

Background:

At an industry meeting on August 8, 1997, the NRC indicated that the potential for, and consequences of, non-design basis events not addressed in the BWRVIP-05 report should be considered. Later, in a Request for Additional Information (RAI) to the BWRVIP, the NRC requested that the BWRVIP evaluate the potential for a non-design basis cold over-pressure transients (Reference 4) and responded to in BWRVIP letter to NRC dated December 18, 1997 (Reference 5). The NRC also considered beyond design basis events, such as low temperature over-pressure (LTOP) events in their PFM analysis. In the BWRVIP response to the RAI the total probability of an occurrence of cold overpressure for BWR-4s was reported as 9E-4.

It was concluded that it is highly unlikely that a BWR would experience a cold over-pressure transient. In fact, for a BWR to experience such an event would generally require several operator errors. The NRC described several types of events that could be precursors to BWR RPV cold over-pressure transients. These were identified as precursors because no cold over-pressure event has occurred at a U.S. BWR. Also, the NRC identified one actual cold over-pressure event that occurred during shutdown at a non-U.S. BWR. This event apparently included several operational errors that resulted in a maximum RPV pressure of 1150 psi with a temperature range of 79°F to 88°F.

For the FitzPatrick plant, the probability for a cold over-pressure event would be lower than that reported above due to the short amount of time that the plant will be in a cold condition over the next cycle compared to that over the license of the plant. The plant is on a 24 month cycle, and does not have a planned outage to a cold condition during the next cycle. The following addresses the high pressure injection sources, administrative controls, and operator training regarding a cold overpressure event for the FitzPatrick plant.

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Review of Potential High Pressure Injection Sources:

The high-pressure make-up systems at FitzPatrick (i.e., the Feedwater, High Pressure Coolant Injection (HPCI), and the Reactor Core Isolation Cooling (RCIC) systems) are steam turbine driven. During reactor cold shutdown conditions, no steam is available for operation of these systems. Therefore, it is not plausible for these systems to contribute to an overpressurization event while the unit is in cold shutdown.

During reactor cold shutdown conditions the condensate booster pumps are normally maintained in the "pull-to-lock" position and the feedwater discharge isolation valves are normally maintained in the closed position. It would require several Operator errors and breakdowns in the work control process to inadvertently start a condensate booster pump and inject into the vessel. As discussed below, operating procedural restrictions, operator training, and work control processes at JAFNPP provide appropriate controls to minimize the potential for RPV cold over-pressurization events.

During normal cold shutdown conditions, RPV level and pressure are controlled with the Control Rod Drive (CRD) and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. The RPV is not taken solid during these items, and plant procedures require opening of the head vent valves after the reactor has been cooled to less than 212°F. If either of these systems were to fail, the Operator would adjust the other system to control level. Under these conditions, the CRD system typically injects water into the reactor at a rate of <60 gpm. This slow injection rate allows the operator sufficient time to react to unanticipated level changes and, thus, significantly reduces the possibility of an event that would result in a violation of the pressure-temperature limits.

The Standby Liquid Control (SLC) system is another high pressure water source to the RPV. However, there are no automatic starts associated with this system. SLC injection requires an Operator to manually start the system from the Control Room or from the local test station. Additionally, the injection rate of the SLC pump is approximately 50 gpm, which would give the Operator ample time to control reactor pressure in the case of an inadvertent injection.

Pressure testing of the RPV is classified as an "Infrequently Performed Test or Evolution" which ensures that these tests receive special management oversight and procedural controls to maintain the plant's level of safety within acceptable limits. The pressure test is conducted so that the required temperature bands for the pressure increases are achieved and maintained prior to increasing pressure. During performance of an RPV pressure test, level and pressure are controlled using the CRD and RWCU systems using a "feed and bleed" process. Increase in pressure is limited to less than 30 psig per minute. This practice minimizes the likelihood of exceeding the pressure-temperature limits during performance of the test.

Procedural Controls/Operator Training to Prevent Reactor Pressure Vessel Cold Over-Pressurization:

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Operating procedural restrictions, operator training, and work control processes at JAFNPP provide appropriate controls to minimize the potential for RPV cold over-pressurization events.

During normal cold shutdown conditions, reactor water level, pressure, and temperature are maintained within established bands in accordance with operating procedures. The Operations procedure governing Control Room activities requires that Control Room Operators frequently monitor for indications and alarms to detect abnormalities as early as possible. This procedure also requires that the Shift Manager be notified immediately of any changes or abnormalities in indications. Furthermore, changes that could affect reactor level, pressure, or temperature can only be performed under the knowledge and direction of the Shift Manager or Control Room Supervisor. Therefore, any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected. Finally, plant conditions and on-going activities that could affect critical plant parameters are discussed at each shift turnover. This ensures that on-coming Operators are cognizant of activities that could adversely affect reactor level, pressure, or temperature.

Procedural controls for reactor temperature, level, and pressure are an integral part of Operator training. Specifically, Operators are trained in methods of controlling water level within specified limits, as well as responding to abnormal water level conditions outside the established limits. Additionally, Control Room Operators receive training on brittle fracture limits and compliance with the Technical Specification pressure-temperature limits curves. Plant-specific procedures have been developed to provide guidance to the Operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

During plant outages the work control processes ensures that the outage schedule and changes to the schedule receive a thorough shutdown risk assessment review to ensure defense-in-depth is maintained. At JAFNPP outage work requests are scheduled by the work control center. Senior Reactor Operators assigned to the work control center provide oversight of outage schedule development to avoid conditions which could adversely impact reactor water level, pressure, or temperature. From the outage schedule, a daily schedule is developed listing the work activities to be performed. These daily schedules are reviewed and approved by Management, and a copy is maintained in the Control Room. Changes to the schedule require Management review and approval.

During outages, work is coordinated through the work control center, which provides an additional level of Operations oversight. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor level or decay heat removal during refueling outages. The Control Room Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Pre-job briefings are conducted for complex work activities, such as RPV pressure tests or hydrostatic testing that have the potential of affecting critical RPV parameters. Pre-job briefings are attended by cognizant individuals involved in the work activity. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

Conclusion

Deferral of the RPV shell weld exams for an operating cycle to ensure greater weld examination coverage will ensure a high degree of quality and safety. Based on the documentation in the BWRVIP-05 report, the risk-informed independent assessment performed by the NRC staff, the lower neutron fluence, the less challenging design and operational loading for BWRs, the quality of the original vessel fabrication, the lack of significant degradation mechanisms, the results of the previous vessel examinations, and controls to prevent a cold over-pressure event, the Authority believes a deferral in completing the inspection of the RPV shell welds until RO-14 provides an acceptable level of quality and safety.

E. ALTERNATIVE EXAMINATIONS:

The JAFNPP alternative plan would require the deferral of the augmented exams to refueling outage 14 (currently scheduled for 4th quarter 2000). The Authority 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 will 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. If the recommendations of the BWRVIP are approved, the Authority will evaluate the planned scope of examinations in relation to conformance with the approved guidance.

References:

1. EPRI TR-105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), September 1995.

2. NYPA letter, J Knubel, to NRC, "Proposed Alternatives in Accordance with 10CFR50.55a(a)(3)(i) for Reactor Pressure Vessel Shell Weld Examinations," (JPN-98-22) dated May 28, 1998.

3. GE Nuclear Energy Accessibility Study, GENE B13-01869-081, Revision 0, June 1997.

4. NRC Letter from Brian W. Sheron, Director, Division of Engineering, Office of Nuclear regulatory Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, "transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information", dated August 14, 1997.

5. BWRVIP Letter, Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, to the NRC, C. E. Carpenter, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-05", dated December 18, 1997.

6. NRC Letter from Gus C. Lainas, Acting Director, Division of Engineering, Office of Nuclear regulatory Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, dated July 28, 1998.

7. NRC Information Notice 97-63, Supplement 1: Status of NRC Staff's Review of BWRVIP-05, dated May 7 1998.

VISUAL INSPECTION RESULTS OF WELDED ATTACHMENTS

NEW YORK POWER AUTHORITY — NDEP: 9.5-6

DATE: 11/07/94

NONDESTRUCTIVE EXAMINATION PROCEDURE — REVISION: 4

ATTACHMENT 4.8

VISUAL EXAMINATION OF THE REACTOR VESSEL INTERNALS AND WELDED ATTACHMENTS

DATA SHEET 9 - VESSEL ATTACHMENT WELDS

Exam Category B-N-2 Direct Visual N/A Remote Visual ✓

Surface Preparation Methods/Tools used NA N/A

Illumination Method used (2) 40 Watt Lights for ROV / (2) 50W Lights for ETV-1250

Direct/Remote Equipment used (ROV) w/RCS 620 ~~w/ RW~~ JB 12/15/94 Camera System / Westinghouse ETV-1250

1. Guide Rod Brackets (2) 0 deg Not Req'd. 180 deg No Rec. Ind.

2. Dryer Support Brackets (4) 4 deg No Rec. Ind. 94 deg No Rec. Ind.
   184 deg No Rec. Ind. 274 deg No Rec Ind.

3. Dryer Hold Down Brackets (4) 41 deg Not Req'd. 138 deg Not Req'd.
   221 deg Not Req'd. 318 deg Not Req'd.

4. Surveillance Sample Holder Wall Brackets (6)

	UPPER	LOWER
30 deg	Pitting and/or Crud Flaking	No Rec. Ind.
120 deg	No Rec. Ind.	No Rec. Ind.
300 deg	Pitting and/or Crud Flaking	No Rec. Ind.

5. Shroud support plate to vessel weld 12/21/94 JHB ~~Not Req'd.~~ H-9 2" each side gussets listed below

6. Shroud gusset to vessel weld 15°, 45°, 75°, 135°, 165°, 195°, 225°, 255°, 315°, 345° No Rec. Ind.

7. Shroud gusset to shroud support plate weld 15° 45° 75° 135° 165 195 225° 255° 315° 345° No Rec. Ind.

Meets ✓ Does Not Meet N/A ASME Section XI Requirements

Examiner [signature] Level II Date 12-7-94 12-8-94, 12-12-94, 12-19-94 12-27-94 Gussets 12-12-94

Reviewer [signature] Level III III III Date 12-22-94 12-21-94

ANII [signature] Date 1-26-95

EXAMINER	LEVEL	DATE
[signature]	III	12/7/94
D. McNeil	II	12/21/94
W. Farrell	II	12/8/94
J. Killoran	II	12-7-94
N/A	N/A	N/A

[signature] Level III W (NYPA) 3-13-95

VISUAL EXAMINATION OF THE REACTOR VESSEL INTERNALS AND WELDED ATTACHMENTS

DATA SHEET 4 - FEEDWATER SYSTEM

Exam Category B-N-1 Direct Visual N/A Remote Visual X

Surface Preparation Methods/Tools used NA N/A

Illumination Method used (2) 90 WATT LIGHTS

Direct/Remote Equipment used RCV w/RCS 620 CAMERA SYSTEM

		LOOP A 45°	LOOP A 315°	LOOP B 135°	LOOP B 225°
1.	Condition of Sparger	NOT REQ'D.	NO REC. IND.	NOT REQ'D.	NO REC. IND.
2.	Nozzle Welds	NOT REQ'D.	(4) FOUR BENT NOZZLES	NOT REQ'D.	NO REC. IND.
3.	Condition of Support Brackets	No Rec. Ind.	NO REC. IND.	No Rec. Ind.	NO REC. IND.
4.	Support Bracket Tack Welds	No Rec. Ind.	NO REC. IND.	No Rec. Ind.	NO REC. IND.
5.	End Brackets & Welds	No Rec. Ind.	NO REC. IND.	No Rec. Ind.	NO REC. IND. *
6.	Junction Box & Welds	NOT REQ'D	N/A	NOT REQ'D.	N/A
7.	Nozzle (Nureg 0619)	No Rec. Ind	NO REC. IND	No Rec. Ind.	NO REC. IND.
8.	Others	N/A	N/A	N/A	N/A

Meets ✓ Does Not Meet N/A ASME Section XI Requirements

Examiner [signatures] Level II / III Date 12-12-94 / 12/13/94

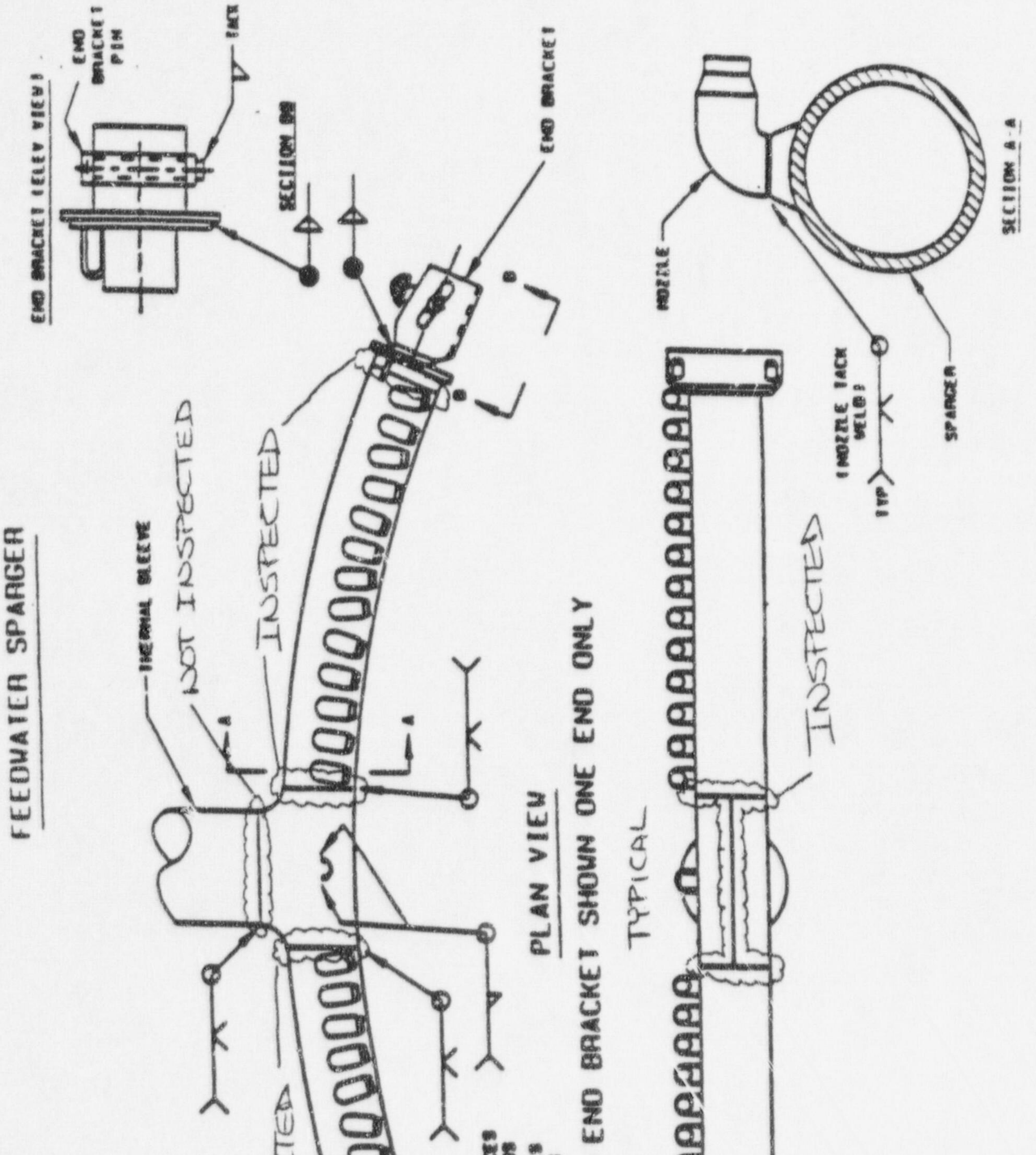
Reviewer [signatures] Level III / III Date 1-4-95 / 12-14-94

ANII [signature] Date 1-26-95

[signature] 5-13-95 Level III VT (NYPA)

* GAP BETWEEN BOLT HEAD AND BRACKET @ 185°
SEE TAPE 94-06, COUNTS 1.03:00

ADDITIONAL EXAMINERS W. FARREL LEVEL II DATE 12-12-94
K HAMMOND II 12-16-94


FEEDWATER SPARGER
THERMAL SLEEVE
NOT INSPECTED
INSPECTED
INSPECTED
PLAN VIEW
END BRACKET SHOWN ONE END ONLY
TYPICAL
INSPECTED
END BRACKET
END BRACKET (ELEV VIEW)
END BRACKET PIN
TACK
SECTION BB
NOZZLE
SPARGER
(NOZZLE TACK WELD)
TYP
SECTION A-A
BOTH ENDS
TWO PLACES BOTH ENDS
TWO PLACES BOTH ENDS

VISUAL EXAMINATION OF THE REACTOR VESSEL INTERNALS AND WELDED ATTACHMENTS

DATA SHEET 5 - CORE SPRAY SYSTEM

Exam Category B-N-1 / B-N-2 Direct Visual N/A Remote Visual ✓

Surface Preparation Methods/Tools used NA N/A

Illumination Method used (2) 40 WATT LIGHTS for PHANTOM / TWO 50W LIGHTS for ETV-1250

Direct/Remote Equipment used RCS-620 (PHANTOM) / WESTINGHOUSE ETV-1250

	Core Spray Piping	LOOP A	LOOP B
1.	"T" Box Welds	NRI	NRI
2.	Support Brackets & Tack Welds	NRI	NRI
3.	Piping to Elbow Welds	NRI	NRI
4.	Piping to Sleeve Welds	NRI	NRI
5.	Downcomer to Pup Piece Weld (includes clamp repair)	NA	NRI
6.	Thermal Sleeve Welds	NRI	NRI
	Sparger		
7.	Junction Box Weld	NRI	NRI
8.	Support Brackets	NRI	NRI
9.	Nozzle Welds	NRI	NRI
10.	End Plug Welds	NRI	NRI

11. Others SCRATCHES, DRAG MARKS ON SPARGER PIPING - NRI

Meets ✓ Does Not Meet N/A NRC IE Bulletin 80-13

Examiner [signature] / M. G. [signature] Level III / II Date 1-5-95 / 12-19-94 (rev) 01-24-95

Reviewer M. G. [signature] / [signature] Level III / III Date 1-5-95 / 1-5-95

ANII [signature] Date 1-26-95

[signature] 3-13-95 Level III (NYPA)

ADDITIONAL EXAMINER K. HAMMOND LEVEL II DATE 12-20-94

GE Nuclear Energy

REACTOR PRESSURE VESSEL INTERNAL VISUAL EXAMINATION DATA SHEET

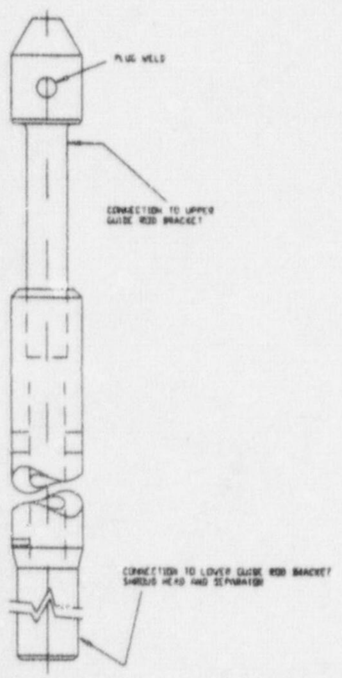
ATTACHMENT #2

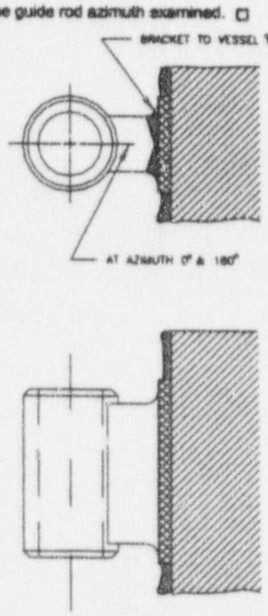
PLANT:	UNIT:	OUTAGE:	IDENTIFICATION OF COMPONENT(S) EXAMINED:
James A. Fitzpatrick Station	1	RFO 12	VT-3 - Guide Rod @ 0°

TECHNIQUE	EQUIPMENT	LIGHTING	TOOLS	RESOLUTION
X REMOTE	X REMOTE B/W VIDEO	X HI INTENSITY	☐ FLOAT BOX	X VT-1, 1/32" LINE
☐ DIRECT	☐ REMOTE COLOR VIDEO	☐ FLASHLIGHT	☐ BINOCULARS	X 0.001" WIRE
	☐ MIRROR	☐ AMBIENT	☐ OTHER	☐ 0.0005" WIRE
	☐ MAGNIFIER	☐ OTHER	N/A	☐ VT-3
	☐ OTHER			☐ OTHER

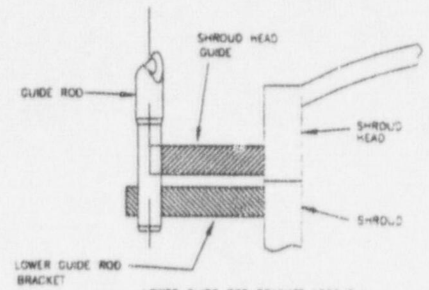
Reference(s)/ Cross Reference(s) TAPE NO. 96-02, 0:00:00 TO 0:29:49, G.E. DRAWING 117C1480 (JAF FILE 5.11-15A)
PROCEDURE # VT-FPK-202V1, REV. 1

EXAMINATION CHECKLIST DESCRIPTION	EXAMINATION RESULTS / REMARKS
Condition of bracket to vessel wall weld. ☐	X NRI
General condition of upper guide rod bracket. ☐	X NRI
Condition of plug weld. ☐	X NRI PLUG WELD VERIFIED PER REF. DRAWING
General condition of guide rod including lower plug and bracket attachment to shroud ☐	X NRI PLUG WELD VERIFIED PER REF. DRAWING MACHINED FLUSH NOT VISIBLE /ACCESSIBLE
Record the guide rod azimuth examined. ☐	X NRI GUIDE ROD @ 0°





UPPER GUIDE ROD BRACKET



LOWER GUIDE ROD BRACKET ASSEMBLY

(UPPER AND LOWER GUIDE ROD BRACKETS NOT THE SAME SCALE)

	NYPA ISI Engineering Review [signature]	Date 11/7/96
Examiner's Signature [signature] J. KILLORAN VT Level II Date 11/3/96	Independent Review Signature Rodney S Barnes R. S. BARNES VT Level III Date 11/5/96	
	NYPA QA Review [signature]	Date 11/7/96
	ANII Review [signature]	Date 11/7/96

Sketches are provided for convenience purposes only and may not represent actual plant configuration and are not to scale.

Attachment 7 - *Feedwater Sparger - 135°*

PLANT:	UNIT:	OUTAGE:	IDENTIFICATION OF COMPONENT(S) EXAMINED:
James A. Fitzpatrick Station	1	RFO 12	VT-3 - Feedwater Sparger @ 135°

TECHNIQUE	EQUIPMENT	LIGHTING	TOOLS	RESOLUTION
X REMOTE ☐ DIRECT	X REMOTE B/W VIDEO X REMOTE COLOR VIDEO ☐ MIRROR ☐ MAGNIFIER ☐ OTHER	X HI INTENSITY ☐ FLASHLIGHT ☐ AMBIENT ☐ OTHER	☐ FLOAT BOX ☐ BINOCULARS ☐ OTHER N/A	X VT-1, 1/32" LINE X 0.001" WIRE X 0.0005" WIRE X VT-3 ☐ OTHER

Reference(s)/ Cross Reference(s) (e.g. Video Tape Id/Counter or VIDS, etc.): TAPE NO. 96-09

EXAMINATION CHECKLIST DESCRIPTION	EXAMINATION RESULTS / REMARKS
General condition of feedwater sparger. ☐	X NRI
Condition of sparger welds. ☐	X NRI
Condition of nozzle welds. ☐	X NRI
Condition of end bracket pins and tack welds. ☐	X NRI NOTE: VT-1 EXAMINATION PERFORMED WITH ITS 1250 HAND HELD CAMERA
Condition of end bracket welds. ☐	X NRI
Record the Feedwater Sparger azimuth examined. ☐	X NRI SPARGER @ 135°

Examiner's Signature	Independent Review Signature		
		NYPA ISI Engineering Review	Date 11/13/96
M. SHARLOW	R. S. BARNES	NYPA QA Review	Date 11/12/96
VT Level II Date 11/09/96	VT Level III Date 11/10/96	ANII Review	Date 11/13/96

Sketches are provided for convenience purposes only and may not represent actual plant configuration and are not to scale.

GE Nuclear Energy

REACTOR PRESSURE VESSEL INTERNAL VISUAL EXAMINATION DATA SHEET

Procedure No. VT-FPK-202V1, Revision1

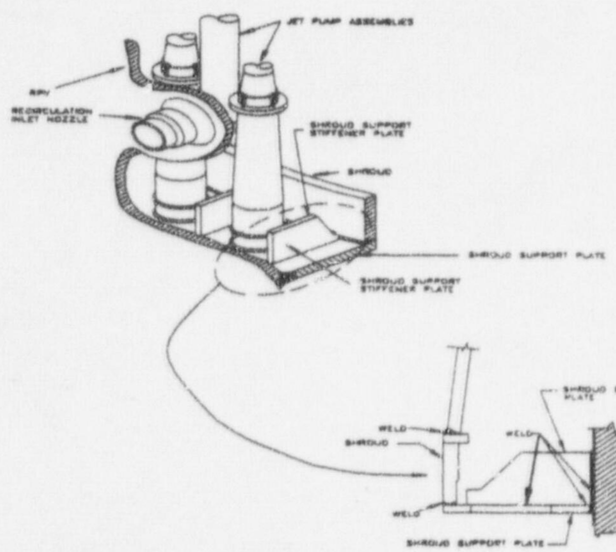
Attachment 14 - Shroud Support Plate Weld

PLANT:	UNIT:	OUTAGE:	IDENTIFICATION OF COMPONENT(S) EXAMINED:
James A. Fitzpatrick Station	1	RFO 12	EVT-1 - Shroud Support Plate Weld Gusset Welds @ 15°, 135°, and 255°

TECHNIQUE	EQUIPMENT	LIGHTING	TOOLS	RESOLUTION
x REMOTE ☐ DIRECT	x REMOTE B/W VIDEO ☐ REMOTE COLOR VIDEO ☐ MIRROR ☐ MAGNIFIER ☐ OTHER	x HI INTENSITY ☐ FLASHLIGHT ☐ AMBIENT ☐ OTHER	☐ FLOAT BOX ☐ BINOCULARS ☐ OTHER N/A	☐ VT-1, 1/32" LINE X 0.001" WIRE X 0.0005" WIRE ☐ VT-3 ☐ OTHER

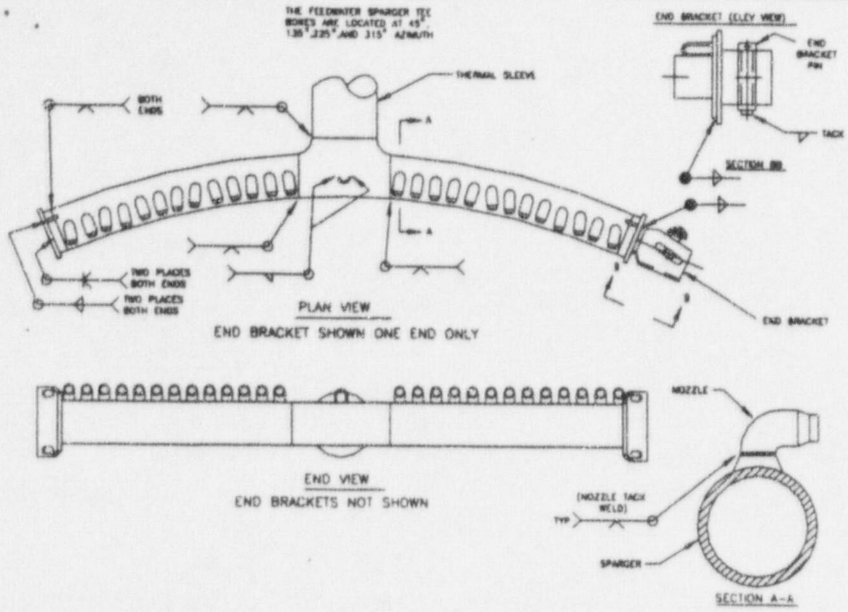
Reference(s)/ Cross Reference(s) (e.g. Video Tape Id/Counter or VIDS, etc.): TAPE NOS. 96-16, 96-17

EXAMINATION CHECKLIST DESCRIPTION	EXAMINATION RESULTS / REMARKS
Condition of accessible shroud support plate weld to reactor vessel. ☐	X NRI SEE NOTE BELOW
Condition of accessible shroud support plate weld to shroud. ☐	☐ NRI N/A
Condition of accessible shroud stiffener (gusset) welds. ☐	X NRI SEE NOTE BELOW
Record the section of shroud support plate weld examined. ☐	X NRI EXAMINED RGV & SGH PORTIONS OF GUSSET (BOTH SIDES) AND 12" EACH SIDE OF GUSSET OF WELD # SH9 AT 15°, 135° AND 255°



Examiner's Signature	Independent Review Signature		
		NYPA ISI Engineering Review	Date 11/15/96
		NYPA QA Review	Date 11/15/96
J. BAZENAS VT Level II Date 11/11/96	R. L. KECK VT Level III Date 11/12/96	ANII Review	Date 11/18/96

Sketches are provided for convenience purposes only and may not represent actual plant configuration and are not to scale.



EXAMINATION OF FEEDWATER SPARGER PERFORMED USING PHANTOM REMOTELY OPERATED VEHICLE. SUPPLEMENTARY VT-1 EXAMINATION PERFORMED ON VESSEL TO BRACKET WELDS AND BRACKET WELDS USING HAND HELD ITS 1250 BLACK AND WHITE CAMERA.

		NYPA ISI Engineering Review	Date 11/13/96
Examiner's Signature	Independent Review Signature	NYPA QA Review	Date 11/13/96
M. SHARLOW	R. S. BARNES	ANII Review	Date 11/13/96
VT Level II Date 11/09/96	VT Level III Date 11/10/96		

Sketches are provided for convenience purposes only and may not represent actual plant configuration and are not to scale.

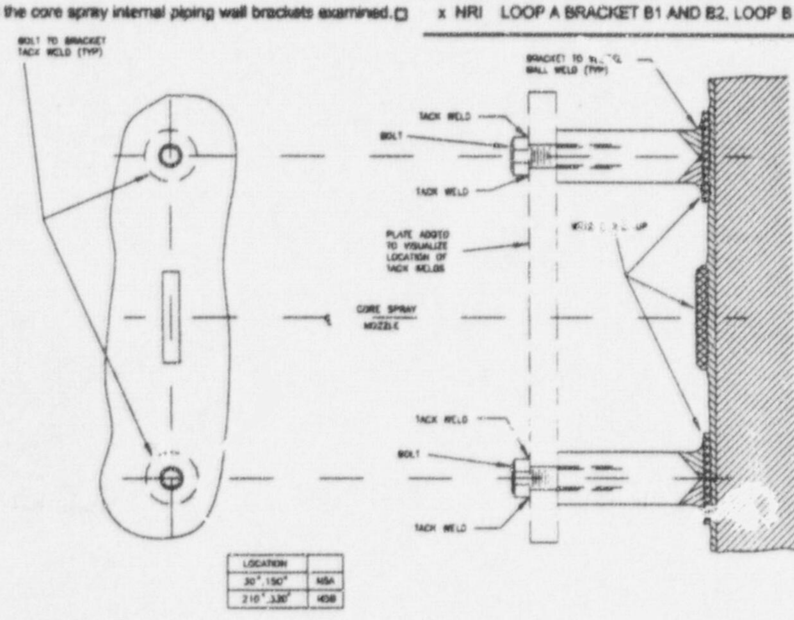
Attachment 9 - *Core Spray Internal Piping Wall Bracket*

PLANT:	UNIT:	OUTAGE:	IDENTIFICATION OF COMPONENT(S) EXAMINED:
James A. Fitzpatrick Station	*1*	*RFO 12*	*EVT-1 - Core Spray Internal Piping Wall Bracket* *Loop A and Loop B*

TECHNIQUE	EQUIPMENT	LIGHTING	TOOLS	RESOLUTION
x REMOTE	x REMOTE B/W VIDEO	x HI INTENSITY	☐ FLOAT BOX	X VT-1, 1/32" LINE
☐ DIRECT	☐ REMOTE COLOR VIDEO	☐ FLASHLIGHT	☐ BINOCULARS	X 0.001" WIRE
	☐ MIRROR	☐ AMBIENT	☐ OTHER	X 0.0005" WIRE
	☐ MAGNIFIER	☐ OTHER	N/A	☐ VT-3
	☐ OTHER			☐ OTHER

Reference(s)/ Cross Reference(s) (e.g. Video Tape Id/Counter or VIDS, etc.): TAPE NOS. 96-01, and 96-04

EXAMINATION CHECKLIST DESCRIPTION	EXAMINATION RESULTS / REMARKS
Condition of bracket to vessel wall welds. ☐	x NRI
Condition of the bolt head tack welds. ☐	x NRI
General condition of the bracket. ☐	x NRI
Record the core spray internal piping wall brackets examined. ☐	x NRI LOOP A BRACKET B1 AND B2, LOOP B BRACKET B1 AND B2



NYPA ISI Engineering Review	Date 11/13/96
NYPA QA Review per telecon for 9 Devine	Date 11/13/96
ANII Review	Date 11/13/96

Examiner's Signature	Independent Review Signature
J. KILLORAN	R. S. BARNES
M. SHARLOW	
VT Level II Date 11/09/96	VT Level III Date 11/10/96

Sketches are provided for convenience purposes only and may not represent actual plant configuration and are not to scale.

ROCKY

LIST OF COMMITMENTS

Commitment No.	Commitment	Due Date
JAFP-98-0316-01	Submit letter to NRC regarding improvements made for performing augmented examination of the RPV vertical shell welds to maximum extent possible. Include details on exam coverage of each weld, specifically detailing vertical weld length coverage in the belt-line region.	12/31/99