Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

February 16, 1996
3F0296-07

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Inservice Inspection Program (ISI) Supplement 1 to Relief Request 95-030

References: A. FPC to NRC letter, 3F0995-03, dated September 22, 1995.
B. NRC to FPC letter, 3N0196-18, dated January 31, 1996

Dear Sir:

Pursuant to 10 CFR 50.55a(g)(5), Florida Power Corporation (FPC) is submitting the attached Relief Request 95-030, Supplement 1. The supplemented information consists of a revision to the Code of Reference previously provided to reflect the correct Code. The correct Code is the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition through Summer 1983 Addenda. Additionally, FPC has revised the "Basis" section of the relief request to incorporate additional data germane to your Request for Additional Information (RAI), Reference B. FPC is also providing FPC responses to the RAI and Structural Integrity Report SIR-96-016 (Attachment 1); and Structural Integrity Report SIR-95-135, "Flaw Acceptance Handbook for Crystal River Unit 3 Reactor Pressure Vessel and Nozzle Weld Inspections" (Attachment 2).

FPC has discussed the changes provided in Relief Request 95-030, Supplement 1 and our response to the RAI with our NRC Project Manager and your contractor, INEL. FPC believes that Supplement 1 to the subject relief request and FPC responses to the questions in the RAI provide a sound basis for the approval of Relief Request 95-030.

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FPC would appreciate your prompt review of that information, in order to implement the request during our refueling outage scheduled to start February 29, 1996.

Sincerely,

Gary Boldt

G. L. Boldt
Vice President
Nuclear Production

GLB/LVC

Attachments
xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager
Michael T. Anderson - INEL Research Center

FLORIDA POWER CORPORATION
INSERVICE INSPECTION
RELIEF REQUEST # 95-030
SUPPLEMENT 1
CRYSTAL RIVER UNIT 3

REFERENCE CODE: ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition through Summer 1983 Addenda

I. COMPONENT FOR WHICH RELIEF IS REQUESTED:

(a) Name and Identification Number:

Reactor Vessel Transition Piece to Bottom Head Weld ISI Exam Number B1.2.2 (see attached sketch)

(b) Function:

Houses Core and Maintains Core Geometry

(c) ASME Section III Code Class:

Class I

(d) Category:

Category B-A, Pressure Retaining Welds in Reactor Vessel

II. REQUIREMENT FROM WHICH RELIEF IS REQUESTED:

ASME Boiler and Pressure Vessel Code, Section XI, Table IWB-2500-1 Item B1.21, Circumferential Head Weld. Volumetric examination of this weld requires the examination of its entire accessible length without allowance for restrictions.

III. BASIS FOR REQUESTING RELIEF:

The subject weld is the reactor vessel transition-piece-to-bottom-head weld. This weld is located below the beltline region and is not subject to the majority of the neutron flux escaping from the core. An evaluation of neutron embrittlement as a potential damage mechanism and other potential damage mechanisms associated with this weld is included in the attached response to Question No. 2A of Reference B (see Attachment 1) and Attachment 2. The evaluation concludes that service-induced degradation of the transition-piece-to-bottom-head weld as a result of corrosion, fatigue, nuclear, or thermal embrittlement mechanisms is extremely unlikely.

The weld has been visually and ultrasonically inspected once during preservice inspection (essentially 100% coverage). The volumetric examination method utilized during the pre-service inspection was Manual Contact Ultrasonic. During this examination the weld received a 360 degree scan with the exception of those areas where physical interference prevented examination with the manual transducer. A review of the data sheets for this examination revealed that there were no reportable or recordable indications detected.

BASIS FOR REQUESTING RELIEF (Continued)

During Refueling Outage 5 (May 1985), the weld was partially inspected (approximately 5%). This inspection was performed per the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition with Addenda through Summer 1975 and Regulatory Guide 1.150. The extent of this examination was acceptable since the 1974 Edition of ASME Section XI, Table IWB 2500, Category B-A only required the examination of 5% of this weld. The examination was performed using the ARIS II remote scanner, a device that utilized immersion ultrasonic techniques. The examination revealed no baseline indications, no reportable indications and no recordable indications.

Although use of the immersion method allowed the weld to be inspected with inspection equipment at a distance of up to 20 inches away from the weld, access for examination of this weld was severely limited by the flow stabilizers, the core support lugs, and the incore instrumentation nozzles.

Since the last inservice inspection was performed, improvements in volumetric examination methods have shown the contact examination method to be much more accurate and reliable than the immersion method. As a result, equipment designed to use the immersion method has been abandoned and modern reactor vessel inspection equipment has been designed to utilize the contact examination method.

Crystal River Unit 3 is currently in the second interval of operation. During the last outage of this interval (March 1996), volumetric examination of reactor vessel welds will be performed using modern automated reactor vessel inspection equipment. Equipment developed for use of the contact method of examination requires more physical access to the surface of the weld than the immersion method previously used. Therefore, the surface of the weld which could be successfully examined using this equipment will be less than that achieved during the first interval.

The response to RAI Question No. 2E, addresses the obstructions located on, and adjacent to this weld which creates an area in which it is very difficult to maneuver the ultrasonic transducer manipulator. The response provides a detailed access study which includes information about FPC's concern for potential impact to the incore instrument nozzles while examining this weld. The nozzles are small and manufactured to close tolerances. If an inadvertent collision were to occur, the nozzles could be severely damaged. A damaged nozzle could prevent the reinsertion of an incore instrument or could require a critical-path in-vessel repair.

FPC response to RAI Question No. 2D addresses the radiation dose potential associated with the examination of this lower reactor vessel weld. The response explains how the current estimated dose for the inspection of the weld (minimal) could potentially increase due to damage of the robotic manipulator due to an impact with the interferences surrounding this weld (One inch protrusion which are remnants of flow stabilizers, guide lugs and incore instrumentation guide tubes).

BASIS FOR REQUESTING RELIEF (Continued)

Access to the subject weld from the vessel exterior presents considerable problems. The region underneath the reactor vessel (below the subject weld) is limited and gained only by passage through a small tunnel leading from outside the biological shield wall to the area directly below the reactor vessel and inside the reactor vessel support skirt. Due to the congested nature of the area (52 incore guide tubes that penetrate the lower head, the lower head insulation and its support structure), a technician would have to access the area and perform a manual examination of the weld. ALARA concerns would prohibit the consideration of such manual external inspection of the weld.

A survey of three ISI service organizations which provide reactor pressure vessel examination services (Framatome, Westinghouse/WesDyne, and Southwest Research) resulted in the determination that no alternate equipment is available which would provide significantly better access to the weld. Our survey also revealed that access limitations for inspecting this weld are common to other PWRs, particularly those with a B&W reactor vessel. The survey also confirmed, based on previous industry experience with limited exams performed on equivalent welds during PWR reactor vessel examinations, that the probability of identifying a service induced flaw in the subject weld would be low.

Accordingly, due to the hardship imposed in implementing this requirement Florida Power Corporation requests relief from examination of the transition-piece-to-bottom-head weld based upon:

The results of evaluating potential damage mechanisms for this weld which revealed a low probability of service-induced degradation due to corrosion, fatigue, nuclear, or thermal embrittlement.

The small possibility of a significant flaw existing in the weld as demonstrated by the results of previous examinations of the transition-piece-to-bottom-head weld which identified no reportable or recordable indications.

The lack of identification of any service-induced flaws in any of the reactor vessel welds.

The severe access limitation addressed in this request and further demonstrated by the access study provided in the response to RAI Question No. 2E.

The critical-path outage time and significant cost required to perform the examination.

The industry-wide lack of inspection equipment that would provide better access for the inspection of the weld.

BASIS FOR REQUESTING RELIEF (Continued)

The structural integrity of the weld as it is addressed in the response to RAI Question No. 2C. The response postulates that the stresses in the lower head region are significantly lower than those of the beltline region and that the fracture toughness capability of the bottom head of the reactor pressure vessel is significantly bounded by the existing Technical Specification P-T curve for the beltline region.

The potential increase in radiation dose due to possible damage/repair of the inspection tools during the examination of the weld.

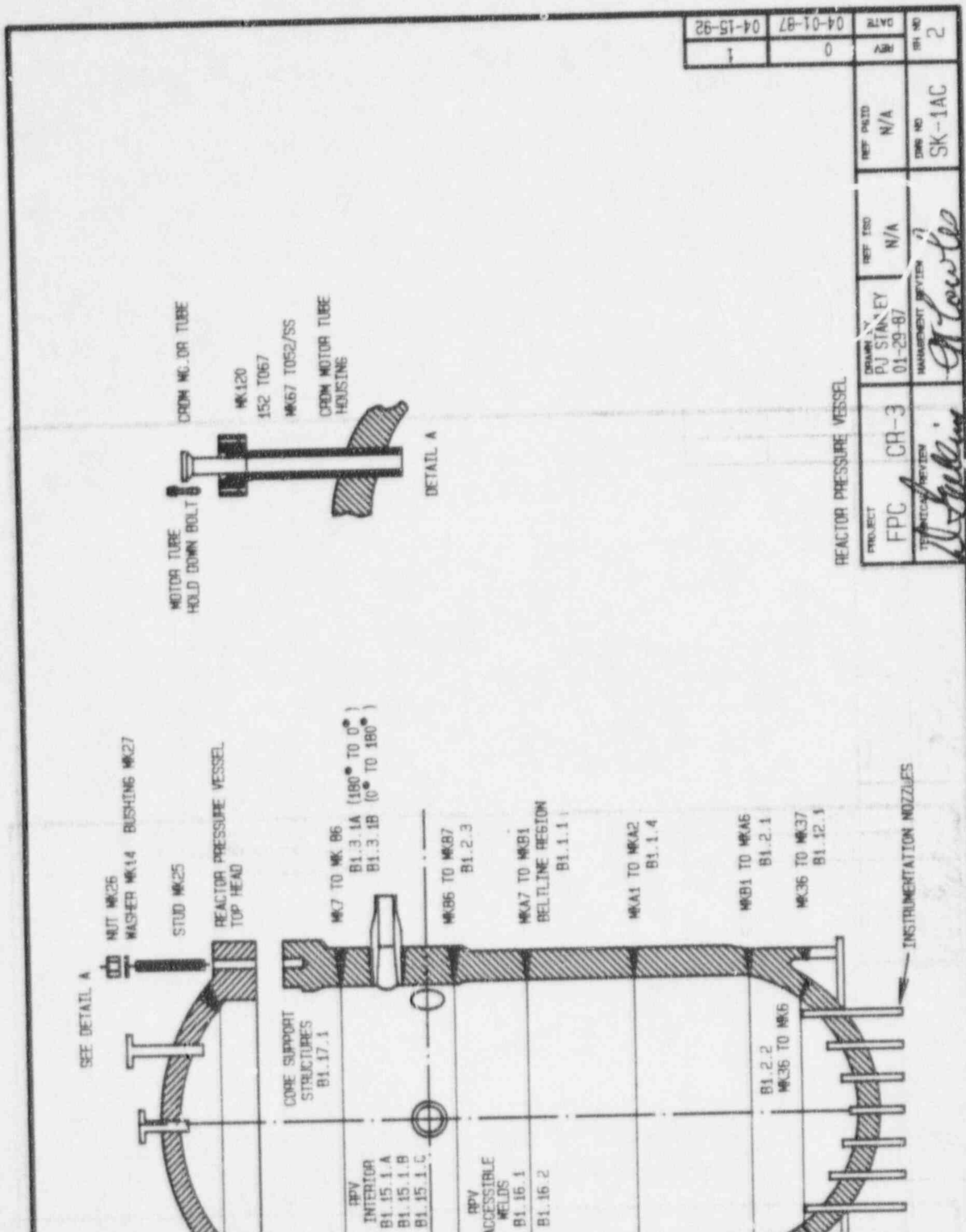
Should examination of the subject weld have to be performed, FPC estimates that the critical-path outage time for performing a limited examination using automated reactor vessel inspection equipment and the contact method would be a minimum of 12 hours, which is estimated to cost approximately $250,000.00.

IV. ALTERNATE EXAMINATION:

The accessible areas of the reactor vessel interior including the interior surfaces and welded attachments within and beyond the beltline region will receive the VT-1 and VT-3 visual examinations required by Section XI of the ASME Code. A VT-2 visual examination will be performed on the exterior of the reactor vessel during the inservice leak test performed during start-up.

V. IMPLEMENTATION SCHEDULE:

The examination of the reactor vessel will be performed during the refueling outage scheduled to begin on February 29, 1996.


SEE DETAIL A
NUT MK26
WASHER MK14 BUSHING MK27
STUD MK25
REACTOR PRESSURE VESSEL
TOP HEAD
MK24 TO MK22
B1.3.2A
B1.3.2B
B1.3.2C
CORE SUPPORT
STRUCTURES
B1.17.1
MK7 TO MK B6
B1.3.1A (180° TO 0°)
B1.3.1B (0° TO 180°)
RPV
INTERIOR
B1.15.1.A
B1.15.1.B
B1.15.1.C
MKB6 TO MKB7
B1.2.3
RPV
ACCESSIBLE
WELDS
B1.16.1
B1.16.2
MKA7 TO MKB1
BELTLINE REGION
B1.1.1
REACTOR PRESSURE
VESSEL
MKA1 TO MKA2
B1.1.4
MKB1 TO MKA6
B1.2.1
B1.2.2
MK36 TO MK6
MK36 TO MK37
B1.12.1
INSTRUMENTATION NOZZLES
MOTOR TUBE
HOLD DOWN BOLT
CRDM MG.OR TUBE
MK120
152 T067
MK67 T052/SS
CRDM MOTOR TUBE
HOUSING
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REACTOR PRESSURE VESSEL
PROJECT
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TECHNICAL REVIEW
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NRC Request for Additional Information

2. A. Discussion of potential damage mechanisms.

The licensee has cited neutron embrittlement as a potential damage mechanism for the shell welds in the beltline region only. For consideration of authorization of this request for relief, the licensee should also address the following:

The reactor pressure vessel transition-piece-to-bottom head weld is of a lesser thickness than the shell welds. Address the stresses and potential damage mechanisms associated with this weld. The discussion should include but not be limited to affects of potential neutron embrittlement on the subject weld (considering the reduced wall thickness), corrosion loads associated with welded attachments (12 flow stabilizer lugs are located on the subject weld), lower head penetrations, expansion/contraction stresses associated with reactor operation cycles and operating conditions.

Response

The information relative to this question is contained in Item A of Structural Integrity Report No. SIR-96-016, attached.

B. Confidence that no flaw is present in the weld.

The licensee has stated that the likelihood of a significant flaw existing in this weld is very small. In the case of the fabrication, preservice, and inservice examinations, the weld was found to be satisfactory. Confirm that there are no preexisting, recordable flaws, acceptable by code.

Response

The lower head (MK-6)to transition piece (MK-36) weld (B1.2.2) received its pre-service examination in August of 1975. The weld was examined by both visual and volumetric examination methods. The volumetric examination method utilized was Manual Contact Ultrasonic and was performed to B&W Ultrasonic (UT) procedures ISI-101 Rev. 8, (Examination of Similar Weld Seams and Attachment Welds) and ISI-102 Rev. 5 (Ultrasonic Examination of the Base Metal Areas Bordering Weld Seams and Base Metal Repairs). During this examination the weld received a 360° scan with the exception of those areas where physical interference prevented examination with the manual transducer. The reactor vessel component that caused the interference are the remnants of the lower flow stabilizers. The flow stabilizers were removed during construction due to problems encountered at another unit while performing hot functional testing. The process left an area of raised metal approximately one inch from the lower head inner surface over the length of the flow stabilizer.

A review of the data sheets for this examination revealed that there were no reportable or recordable indications found. This information was gathered from the Preoperational Inspection Summary Report, dated 10/14/76.

The weld was again examined at the conclusion of the first 10 year interval during Refueling Outage 5 (5/85), as part of the ASME Section XI code required reactor vessel examination. The ultrasonic examination of vessel welds was performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition with Addenda through Summer 1975 and NRC Regulatory Guide 1.150. These examinations were performed by B&W using the ARIS II remote scanner, a device that employed immersion ultrasonic techniques. The extent of examination for this weld was limited to 5 % due to interferences with the guide lugs, incore instrument guide tubes, and the remnants of the lower flow stabilizers. This was acceptable in terms of extent of coverage based on the requirements of the 1974 edition of ASME Section XI, Table IWB-2500, Category B-A which only required 5 % coverage of this weld.

The reactor vessel weld examination summary report documents that there were no baseline indications, no reportable indications, and no recordable indications.

C. Structural Integrity.

The licensee essentially proposes the elimination of the subject volumetric Code examination of the accessible portions of the weld. This implies that other RPV welds are more susceptible to failure than the subject weld. Based on a qualitative comparison of the fracture toughness of the beltline weld to the lower head weld, what is the estimated critical flaw size for the lower head weld (Appendix G ASME Code flaw size.)?

Response

The information relative to this question is contained in Structural Integrity Report No. SIR-96-016, attached. The information regarding critical flaw size is contained in Attachment 2, Structural Integrity Report No. SIR-95-135. Attachment 2 provides flaw acceptance guidelines for Crystal River Unit 3. The applicable guidelines for the subject weld are included in Appendices H,I and J of the attachment. The graphical guidelines are conservative and present flaw acceptance criteria for each of the particular vessel regions (Section 3.0 describes the regions). These charts show a maximum allowable flaw size which is dependent on flaw orientation, aspect ratio and on the material thickness.

D. Radiation Fields.

The licensee has not addressed the radiation dose potential associated with the examination of the subject weld. Provide information on the estimated exposure associated with the examination of the subject weld.

Response

The estimated radiation dose to personnel in performing an examination of this lower reactor vessel weld using the new Framatome robotic manipulator (URSULA) is minimal. The current technology used to perform the ASME Section XI reactor vessel second 10 year welds examination requires minimal personnel working over the reactor vessel. URSULA, once set-up in the vessel, can scan a large portion of the vessel from one location with minimal personnel support. The manipulator is controlled from a remote operations center located outside the reactor building thus minimizing dose. Dose is only accumulated when personnel are required to relocate the unit or to effect equipment repairs should they be required. Since damage is a credible event as a result of attempting to examine this weld, the resulting equipment repairs would increase personnel dose exposure.

The URSULA robotic manipulator is designed to examine the reactor vessel welds from the interior of the reactor vessel. To examine the weld from the vessel exterior presents considerable problems. Access to the region underneath the reactor vessel is limited and gained only by passage through a small tunnel leading from outside the biological shield wall to the area directly below the reactor vessel and inside the reactor vessel support skirt. This area is extremely congested with the 52 incore guide tubes that penetrate the lower head, the lower head insulation, and its support structure. The lower head to transition piece weld is located above all of these interferences at the crotch location of the lower transition piece (MK-36), see the attached B&W drawings 15543 and 126951. Due to the congested nature of the area and the lack of automated equipment to scan the weld, a technician would have to access the area and perform a manual examination of the weld. It is estimated that the total dose exposure accumulated to perform this exam would be 25-36 person-REM with an associated cost of between $250,000.00 and $350,000.00.

E. Potential for Damage Caused by Examinations.

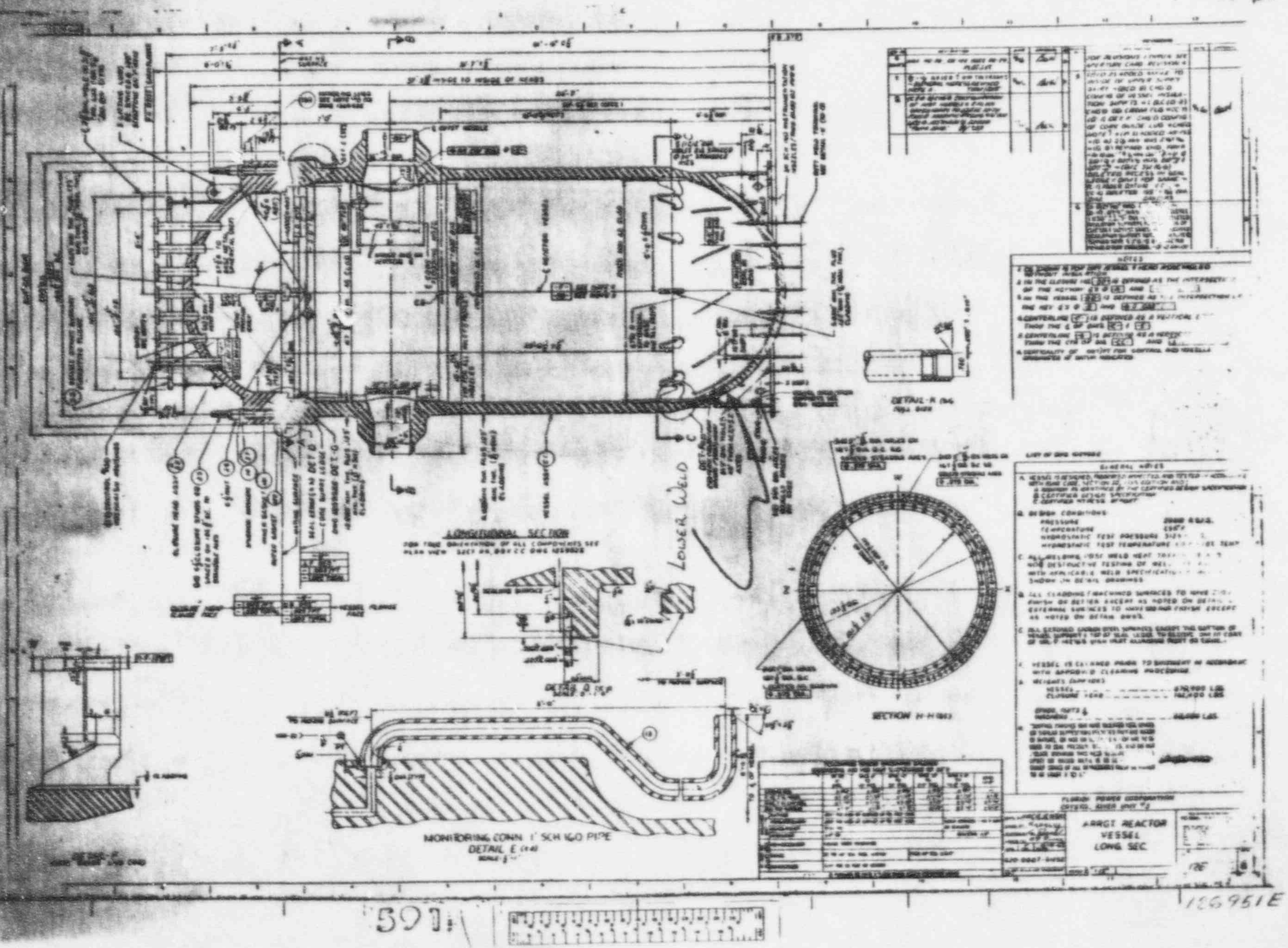
The licensee cites limited access for examination and the potential for damage of incore instrumentation by the examination tool. Provide a detailed access study and determine the actual probability for potential damage due to the inspection tooling,)i.e. considering clearance requirements, tool operations, etc.). In addition, provide instances where damage, if any associated with the subject weld has occurred, the result of the use of the inspection tool at your plant or at any other plant with similar reactor pressure vessel designs.

Response

The Framatome robotic manipulator (URSULA) employees a two by two matrix of contact type ultrasonic transducers in the scanning head. In order to scan a weld, the transducer shoe must be in contact with the vessel surface. As can be seen from the attached drawings (B&W Dwg #'s 126951 and 135543) the area surrounding the lower head to transition piece weld is extremely congested with other reactor vessel components, such as the guide lugs, the incore instrumentation guide tubes, and the remnants of the lower flow stabilizers. The reactor vessel component of immediate concern, should an impact occur with the URSULA scanning head, are the incore instrument guide tubes. These slender tubes protrude from the inner surface of the lower head approximately 12 inches and are less than 1 inch in diameter. Alignment of these tubes is critical as they are inserted into a mating tube on the lower section of the core barrel when it is replaced. Any misalignment of these tubes due to an impact with the scanning head would cause significant damage not only to the incore guide tube, but also to the mating tube on the core support assembly and any adjacent tubes if significant deflection were to occur while installing the core barrel. Repair to the damaged incore instrumentation guide tube and any other affect components would cause significant critical path delays to the outage schedule.

Another issue to be contended with is the potential for damage to the URSULA manipulator arm due to an impact with one or more of the obstructions in the vicinity of the weld. This is significant from the stand point of accurate transducer positioning. If damage were to occur to the arm such that it is unable to calculate its position relative to the base unit, accurate positioning of the UT data being collected would be lost. To recover from this situation would entail repair of the manipulator arm and a rescan of the weld; both evolutions adding to the critical path time of the schedule and dose exposure for the job.

The first time deployment of the URSULA manipulator to a B&W reactor vessel was at ENTERGY's ANO-1. While developing the scan plan for the vessel inspection, it was determined that access to the lower head to transition piece weld would be limited to less than 7 %. The scan plan was developed using 3D CAD software (ROBOCAD) which incorporates the actual vessel dimensions and all the interferences. This allowed the programmers to determine how much of the weld would actually be accessible for scanning.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LOWER WELD
MONITORING CONN. 1" SCH 160 PIPE
DETAIL E
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DETAIL-R
GENERAL NOTES
NOTES
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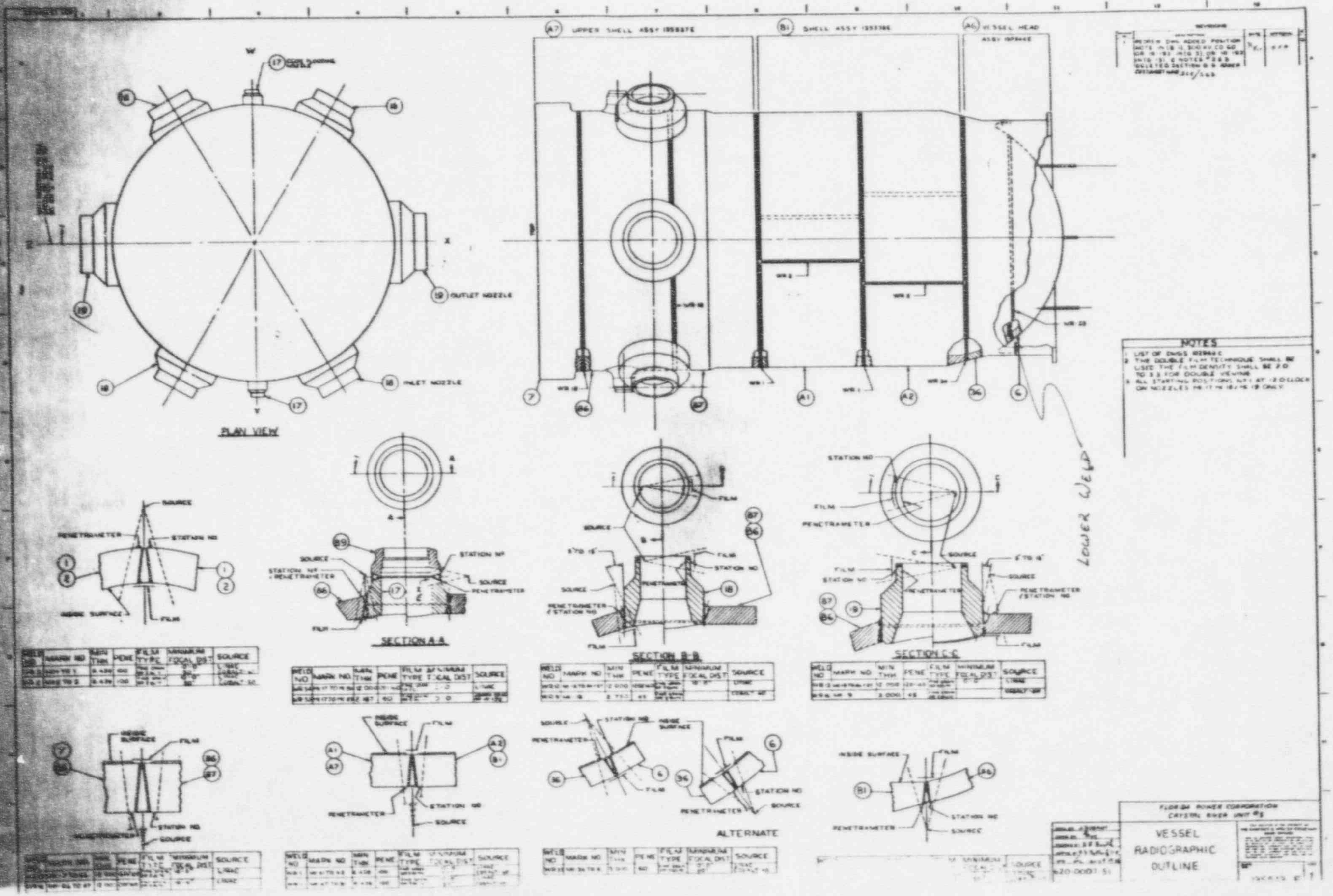
UPPER SHELL ASSY
SHELL ASSY
VESSEL HEAD
PLAN VIEW
INLET NOZZLE
OUTLET NOZZLE
NOTES
SECTION A-A
SECTION B-B
SECTION C-C
SOURCE
PENETRAMETER
STATION NO
FILM
INSIDE SURFACE
ALTERNATE
LOWER WELD
FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT #3
VESSEL
RADIOGRAPHIC
OUTLINE