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

March 12, 1996
3F0396-08

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

Subject: Inservice Inspection Program (ISI) Relief Request 95-030, Supplement 2

References: A. NRC to FPC letter, 3N0196-18, dated January 31, 1996
B. FPC to NRC letter, 3F0296-07, dated February 16, 1996

Dear Sir:

Pursuant to 10 CFR 50.55a(g)(5), Florida Power Corporation (FPC) is submitting the attached Relief Request 95-030, Supplement 2. Supplement 1 provided FPC's responses to your request for additional information (RAI), Reference A. FPC's response to the RAI, Reference B, Question 2C, stated that stresses were lower in the lower head region, but did not clarify which stresses were considered. Analysis of reactor vessel stress has found that while the axial stresses are higher at the lower head region than the axial stresses in the beltline region, the hoop stresses are significantly lower. In addition, the axial stresses at the lower head weld are less than the hoop stresses in the beltline region. The lower stresses at the lower head region combined with its higher available fracture toughness and location in a region of lower neutron fluence gives the lower head region a higher flaw tolerance than the beltline region of the vessel. In addition to this clarification, FPC is revising the response to RAI, Question 2C provided in Attachment 1 of Reference B. The original response to RAI, Question 2C gave an incorrect reference to Attachment 2, Structural Integrity Report NO. SIR-95-135. Attachment 2 provided an analysis in accordance with ASME, Section XI, Appendix A, which requires that cladding and residual stresses be included. RAI Question 2C requested information pertaining to critical flaw size in accordance with ASME, Section III, Appendix G. Information pertaining to Appendix G was actually contained in Attachment 1 to Reference B, Structural Integrity Report NO. SIR-96-016, Pages 7 and 8. FPC is deleting the Reference to Attachment 2, Structural Integrity Report NO. SIR-95-135 from the response to RAI, Question 2C and replacing the reference with results of the Appendix G analysis contained in Structural Integrity Report NO. SIR-96-016.

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FPC would appreciate your prompt review of this information, in order to implement the request during our refueling outage which began February 16, 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 2
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). 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 (started February 16, 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 greater flaw tolerance of the weld as it is addressed in the response to RAI Question No. 2C.

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

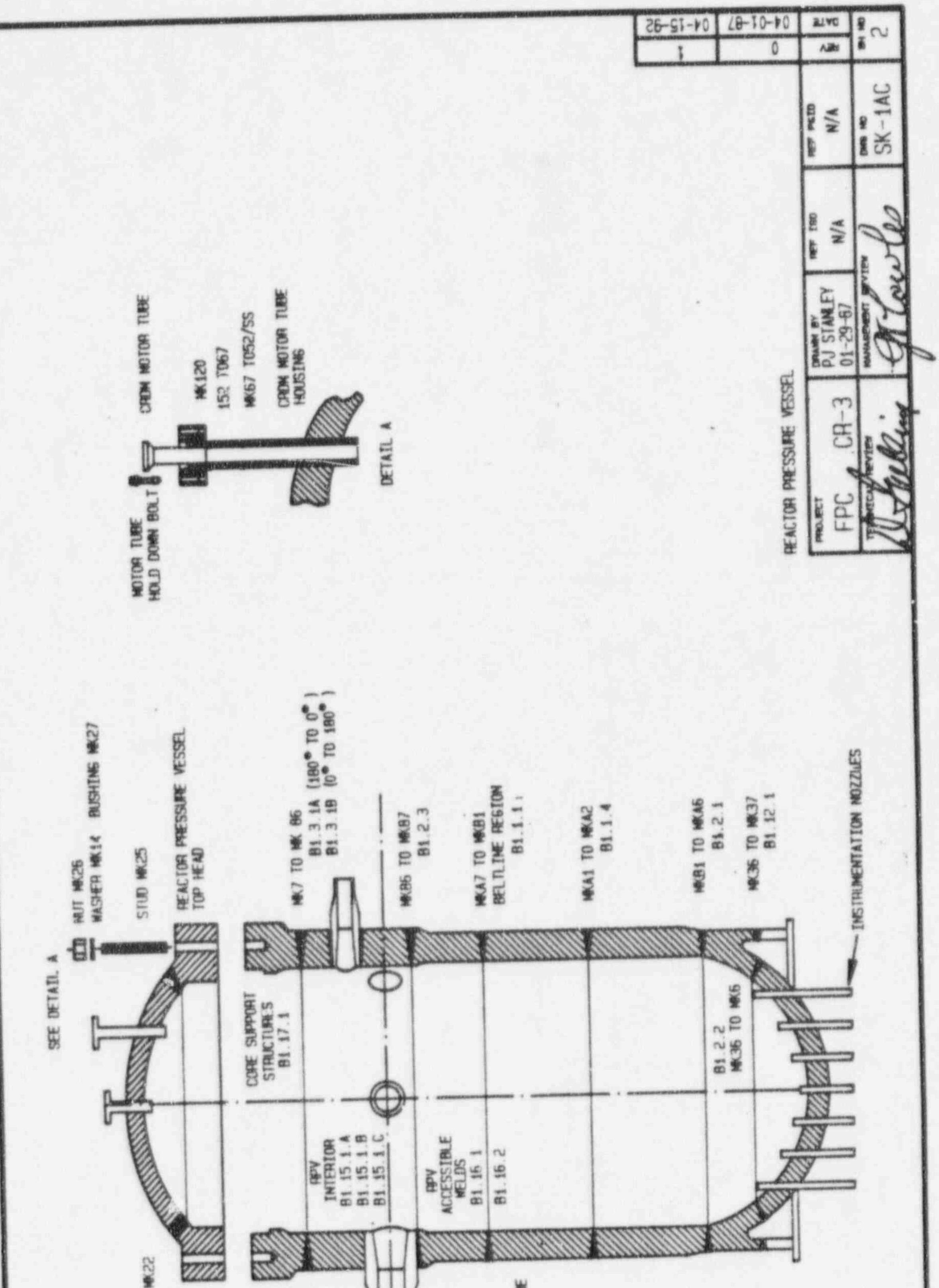
Should examination of the subject weld have to be performed, FPC estimates that the critical-path outage time for performing a limited examination using automated reactor vessel inspection equipment and the contact method would be a minimum of 12 hours, which is estimated to cost approximately $250,000.00.

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 10 which began on February 16, 1996.


MK24 TO MK22
B1.3 2A
B1.3 2B
B1.3 2C
REACTOR PRESSURE VESSEL
SEE DETAIL A
NUT MK26
WASHER MK14 BUSHING MK27
STUD MK25
REACTOR PRESSURE VESSEL TOP HEAD
CORE SUPPORT STRUCTURES B1.17.1
RPV INTERIOR
B1.15.1.A
B1.15.1.B
B1.15.1.C
RPV ACCESSIBLE WELDS
B1.16.1
B1.16.2
MK7 TO MK 86
B1.3.1A (180° TO 0°)
B1.3.1B (0° TO 180°)
MK86 TO MK87
B1.2.3
MKA7 TO MKB1
BELTLINE REGION
B1.1.1
MKA1 TO MKA2
B1.1.4
MKB1 TO MKA6
B1.2.1
MK36 TO MK37
B1.12.1
B1.2.2
MK36 TO MK6
INSTRUMENTATION NOZZLES
MOTOR TUBE HOLD DOWN BOLT
CRDM MOTOR TUBE
MK120
152 T067
MK67 T052/SS
CRDM MOTOR TUBE HOUSING
DETAIL A
REACTOR PRESSURE VESSEL
PROJECT FPC CR-3
DRAWN BY PJ STANLEY 01-29-87
REF ISO N/A
REF P&ID N/A
DWG NO SK-1AC
SH NO 2
REV 0 DATE 04-01-87
REV 1 DATE 04-15-92
TECHNICAL REVIEW
MANAGEMENT REVIEW

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.

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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

FPC's reactor vessel flaw handbook has recently been updated by Structural Integrity Associates, Inc. to support CR-3's ten year reactor vessel inspection. Because the flaw handbook analysis was performed in accordance with ASME, Section XI, Appendix A, which requires that cladding and residual stresses be included, a second analysis was performed in order to assess the estimated critical flaw size in the lower head weld in accordance with ASME, Section III, Appendix G. The results of the Appendix G analysis are contained in Structural Integrity Report NO. SIR-96-016, Pages 7 and 8, attached.

For performing an evaluation in accordance with Appendix G, a defect equivalent to 1/4 of the thickness with a depth-to-length ratio of 1/6 is postulated as the critical flaw size. Analysis is then conducted for Service Level A and B conditions and for hydrotest conditions to determine the allowable pressure/temperature (P-T) operating conditions for the reactor vessel such that the applied stress intensity factor (with required margins) is equal to the Reference Critical Stress Intensity Factor of Appendix G, Figure G-2210-1. Because of high irradiation and high hoop stresses in the reactor beltline region, Appendix G postulated axial flaws are generally controlling in determining P-T Curves.

The critical flaw size in the bottom head region is typically larger than the critical flaw size in the beltline region because the bottom head, unlike the beltline, does not experience significant irradiation effects from exposure to neutron fluence. Although the thickness of the bottom head is smaller than the beltline region, the maximum stresses due to internal pressure are less in the bottom head because the smaller thickness is compensated by the spherical shape (as opposed to the cylindrical shap of the beltline where high hoop stresses exist). For the bottom head wel in question, there are no penetrations or appurtenances immediately adjacen to the weld that would necessitate the use of a stress concentration factor in determining the stresses. Thus, with lower hoop stresses and a higher available toughness, the flaw tolerance capabilities of the bottom head region are expected to be significantly higher than the beltline region.

The Appendix G analysis established curves of allowable pressure versus fluid temperature for various postulated flaw sizes. These analyses utilized the limiting reference temperature for all plate and weld material in the bottom head region. Three flaw sizes were evaluated: 1/4T, 3/8T, and 1/2T. Various fluid temperature heatup and cooldown rates ranging from that identified on the current P-T Technical Specification curve up to 100 degree F/hr were evaluated with respect to both outside surface and inside surface flaws. The limiting 100 degree F/hr results were evaluated for all three flaw sizes and compared against the currently established Technical Specification P-T curve based on the limiting beltline region, for the hydrotest condition. The results of the analysis demonstrated that the current P-T curve for the limiting beltline region significantly bounds that of the lower head region. For flaw sizes as large as 1/2 T, where T is the thickness of the subject weld (5-3/8"), significant margin remains. The primary reason for the significant difference between the bottom head and beltline region is a result of decreased beltline toughness caused by irradiation.

Thus, this analysis shows that the critical flaw size for an Appendix G evaluation of the bottom head would be in excess of 1/2 T, more than twice the maximum postulated defect size required by Appendix G.

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.

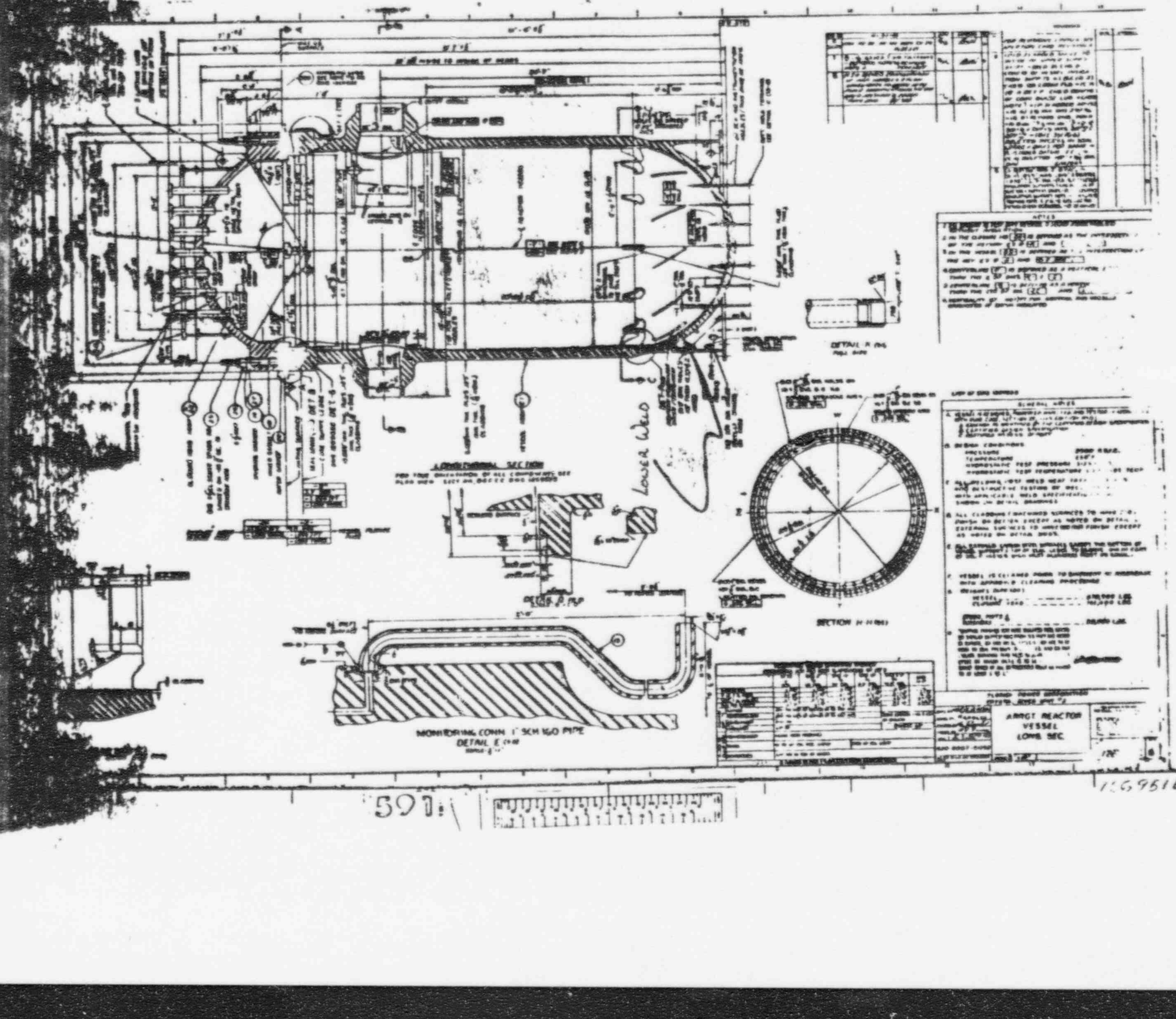
The licensee cites limited access for examination and the potential for damage of incore instrumentation by the examination tool. Provide a detailed access study and determine the actual probability for potential damage due to the inspection tooling,)i.e. considering clearance requirements, tool operations, 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.

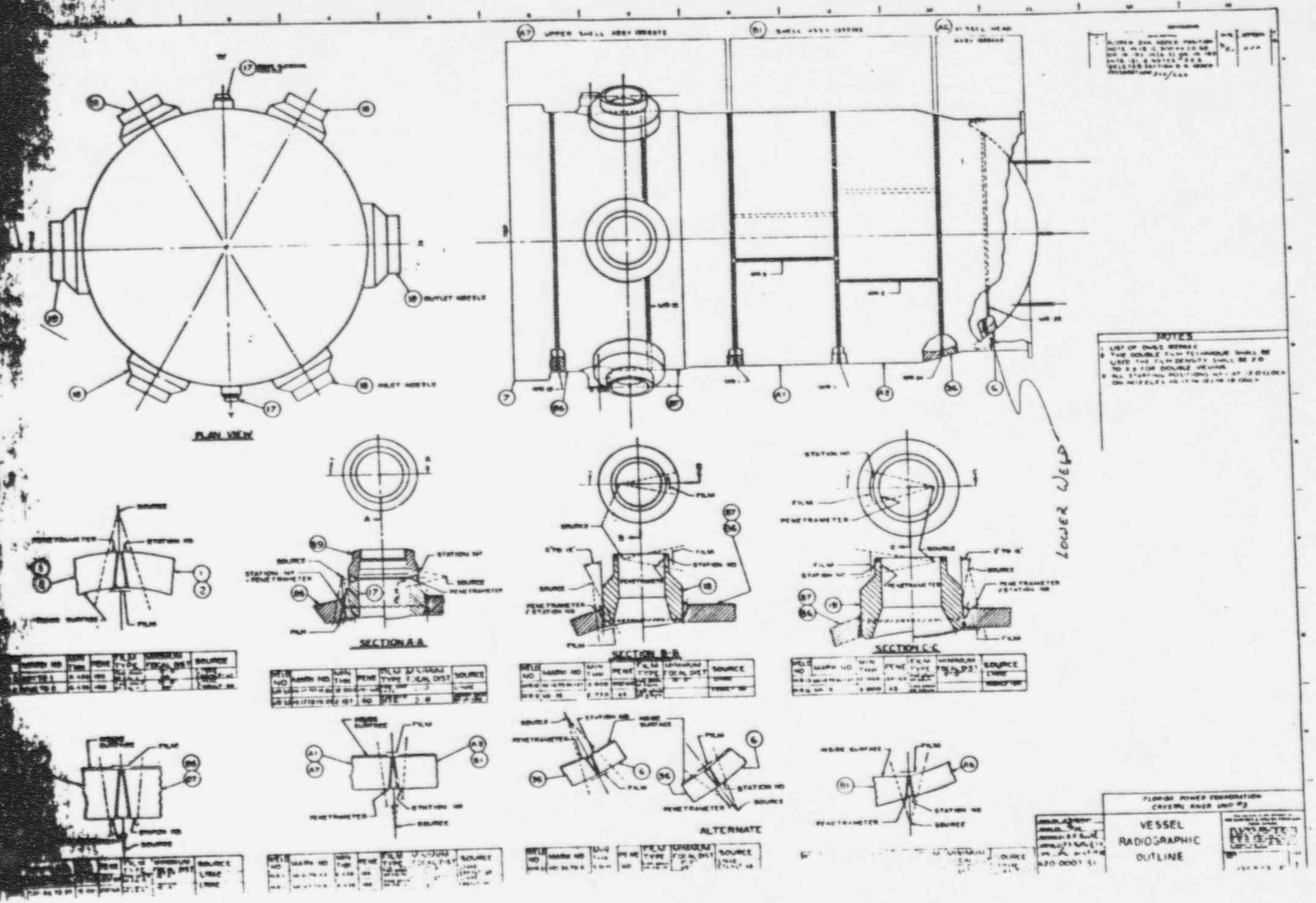
Response

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.


LONGITUDINAL SECTION
LOWER WELD
MONITORING CONN. 1" SCH 160 PIPE
DETAIL E
VESSEL
LONG. SEC.



PLAN VIEW
OUTLET NOZZLE
INLET NOZZLE
UPPER SHELL ASSY
SHELL ASSY
VESSEL HEAD
LOWER WELD
NOTES
SECTION A-A
SECTION B-B
SECTION C-C
SOURCE
FILM
PENETRAMETER
STATION NO
INSIDE SURFACE
ALTERNATE
FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT #3
VESSEL
RADIOGRAPHIC
OUTLINE

U. S. Nuclear Regulatory Commission
3F0296-07

Attachment 1

Report No.: SIR-96-016
Revision No.: 0
Project No.: FPC-02Q
File No.: FPC-02Q-401
February 1996

Input to Items A and C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld at Crystal River Unit 3

Prepared for:

Florida Power Corporation

Prepared by:

Structural Integrity Associates, Inc.
San Jose, CA

~~9602230161~~

Structural Integrity Associates, Inc.

ITEM A

1. EVALUATION OF POTENTIAL DEGRADATION MECHANISMS

The transition piece to bottom head weld is subjected to operation at an elevated temperature of approximately 575°F for long times (on the order of 200,000 hours). The potential effects of thermal aging, irradiation embrittlement, and corrosion must be considered as potential degradation mechanisms at the bottom head weld at Crystal River Unit 3. The change in section from the adjacent transition piece in this area may lead to increased stresses, making it susceptible to thermal fatigue. Finally, the impact of fabrication-related defects must be evaluated.

Corrosion

As noted in EPRI report NP-5461 [1], potential corrosion mechanisms include:

- General corrosion
- Stress corrosion cracking (SCC)
  - Intergranular (IGSCC)
  - Transgranular (TGSCC)
  - Irradiation assisted (IASCC)
- Erosion-corrosion
- Crevice corrosion
- Pitting
- Intergranular attack
- Hydrogen embrittlement
- Microbiologically influenced corrosion (MIC)

The entire inside surface of the reactor vessel is clad with stainless steel with nominal composition and properties equivalent to those of Type 304 [2]. The purpose of the cladding is to provide corrosion protection to the low alloy steel plate and weld metal and to assist in maintaining reactor coolant water purity. The general corrosion rate of the stainless steel in a PWR environment is extremely low. Thus, the contributions of corrosion products to degradation of fuel rod heat transfer, to turbidity of the water, or to acting as a source of activated products that might contribute to dose rates are minimized. The stainless steel is also extremely resistant to pitting or crevice corrosion in the controlled purity PWR coolant. While stainless steels have been shown to be subject to SCC in Boiling Water Reactors, the resistance of stainless steel weld metal to SCC is much greater [3]. Further, the susceptibility of the reactor pressure vessel base material to growth of SCC is extremely low, especially under the non-oxidizing conditions in the PWR reactor vessel. In the extremely unlikely event that reactor water does contact the low alloy steel corrosion degradation would be limited to general corrosion or galvanic corrosion in the low alloy steel at rates much less than 1 mpy [4]. The presence of stainless steel cladding, a material with extremely high resistance to velocity effects, will preclude erosion-corrosion.

Hydrogen embrittlement most often occurs when materials are charged with hydrogen as a result of processing (e.g., plating operations), exposed to a service environment with a high partial pressure of hydrogen and elevated temperature (e.g., refinery vessels), or as a result of hydrogen produced from corrosion. Environmentally assisted cracking may be the result of SCC, the active interaction between stresses and corrosion processes, or hydrogen embrittlement, where hydrogen produced from the corrosion process embrittles the structure at the crack tip producing a brittle type failure. The latter mechanism may be considered to produce the same level of degradation as that described for SCC in the paragraphs above (i.e., degradation due to SCC or hydrogen embrittlement is considered to be unlikely for this location). Since the amount of corrosion that occurs at the cladding/coolant interface is so small, very little hydrogen will diffuse through the stainless steel to the low alloy steel.

The presence of stainless steel cladding effectively eliminates corrosion as a degradation mechanism of the low alloy steel structural components.

Neutron Embrittlement

As shown in Table 1, for the head transition piece, a fluence of 4.79E16 n/cm^2 has been reported [5]. For purposes of this evaluation, this fluence has also been conservatively assumed for the lower shell to head transition weld (WF-154). Using the methodology provided in Reg. Guide 1.99, Rev. 2, this corresponds to a fluence factor of 0.052, resulting in a reference shift of 6.9°F and adjusted reference temperature (ART) of 16.9F°. This reduction in fluence results in a shift which is significantly less than what is reported for the beltline materials, as can be seen in Table 1.

Thermal Embrittlement

Ferritic materials, including pressure vessel steels such as those used for the Crystal River Unit 3 reactor pressure vessel, can be subject to a loss of toughness as a result of long term exposure to elevated temperatures. Temper embrittlement, produced by the segregation of impurities such as arsenic, antimony, and tin to grain boundaries is a strong function of temperature. The effect is observed in the range 750°F to 900°F. The exposure temperature for the Crystal River Unit 3 pressure vessel (as for other LWR reactor vessels) is far too low to produce thermal embrittlement.

Fatigue

As noted above, the change in section at the transition piece to bottom head area makes it a potential area of susceptibility for thermal fatigue. However, the following discussion of stresses demonstrate that this vessel location will not be subjected to transient events, either in large numbers or of a magnitude that will produce thermal fatigue. Fatigue crack growth will be contributed mainly by startup/shutdown transients. Crack growth is expected to be very small for the remaining 167 startup/shutdown cycles at Crystal River Unit 3 [6].

Fabrication Defects

With respect to Crystal River Unit 3, both preservice, as well as the first 10-year ISI inspections, did not identify any reported flaws. As such, in the unlikely event that fabrication-related defects are found, they are expected to be relatively small and insignificant.

Conclusions

This evaluation has demonstrated that service-induced degradation of the transition piece to bottom head weld as a result of corrosion, fatigue, or thermal embrittlement mechanisms is extremely unlikely. The primary contributor to the presence of flaws in that weld is due to fabrication. Prior vessel inspections did not identify any flaws and as such the existence of fabrication related defects in the bottom head weld is unlikely. Relief from inspection of the transition piece to bottom head weld in the Crystal River Unit 3 reactor vessel appears to be justified.

2. STRESSES AND LOADS

Stresses acting on the transition piece to bottom head weld consist of the following:

- Pressure
- Thermal Transients
- Expansion/Contraction Stresses of Cladding
- Loads Associated with Welded Attachments (Flow Stabilizer Lugs)
- Stresses Resulting from Bottom Head Penetrations

The significance of these loads in relation to the weld location are discussed below.

Pressure

An analysis due to internal pressure of 2240 psig was performed in Reference 7. The geometrical details of the reactor pressure vessel shown in Figure 1 were used to construct the finite element model shown in Figure 2. The resulting through-wall pressure stresses in the transition piece to bottom head weld location are shown in Table 2.

Thermal Transients

Several transients are described in the original Reactor Vessel Stress Reports [2]. However, it was determined that the most significant transients in the bottom head region are those associated with plant heatup and cooldown. Analyses were performed using the finite element model shown in Figure 2 to determine the stresses and temperatures associated with the heatup and cooldown transients. Maximum through-wall thermal stresses are shown in Table 2. These stresses are relatively small compared to the pressure stresses. Due to the adjacent thickness change of the transition piece, the thermal stress distributions are altered from those expected for a shell remote of a discontinuity.

Expansion and Contraction Stresses

The expansion and contraction stresses result from differences in the coefficient of thermal expansion between the stainless steel clad and the low alloy pressure vessel steel. As required by the ASME Code, during construction, post weld heat treatment (PWHT) was performed on the vessel at Crystal River Unit 3 following welding operations, including the application of the cladding. PWHT is intended to reduce residual stresses arising from welding. This was achieved by subjecting the entire vessel to a temperature of about 1100° - 1200°F for 12 to 48 hours and then gradually cooling it uniformly to room temperature.

It is generally believed that due to PWHT, the vessel is stress free at the operating temperature of approximately 575°F [8]. However during cooling to room temperature, tensile residual stresses are generated in the stainless steel cladding because stainless steel has a higher coefficient of thermal expansion than the low alloy pressure vessel steel. It has been established that the tensile stress in the stainless steel at room temperature is on the order of yield stress (30-35 ksi for stainless steel) [9]. The cladding stress is compensated by an essentially uniform compressive force in the low alloy base metal. The postulated distribution of the clad stress distribution at room temperature is shown in Figure 3 for the 5" thick bottom head. Because of this stress distribution, the associated stress intensity factor tends to have a very sharp exponential decay through the wall of the vessel for inside surface cracks. Since the stress in the base metal is compressive, cladding induced stresses are not a factor for embedded flaws, the type of flaw most likely to be encountered during inspection. At normal operating temperature (575°F), the effects of cladding stresses diminish due to the relative coefficient of thermal expansion. It is predicted that the cladding-induced tensile stress reduces to a value of approximately 5 ksi at the normal operating temperature, with a similar percent reduction in the base metal.

Loads Associated with Welded Attachments (Flow Stabilizers)

In the original design, there are twelve flow stabilizer plates welded to the lower head in the vicinity of the transition piece to lower head weld with a size of approximately 37 inches by 10 inches. Before plant operation, these plates were cut-out and leaving only 1 inch of the original 10 inch width protruding from the surface of the cladding. As such, the mechanical loads on these lugs are insignificant. The plates were made from stainless steel and as such there is no differential thermal expansion stress between the plate and the stainless steel cladding. The location of these plates extend 1 inch below the centerline of the weld.

Loads from Bottom Head Penetrations

The loads on the bottom head due to the fifty-six (56) 3/4-inch penetrations were analyzed in the original Stress Report [10]. These penetrations are at least 10 inches away from the transition piece to bottom head weld and, as such, the weld location is unaffected by the loads from these penetrations.

ITEM C

1. FRACTURE TOUGHNESS AND ASME CODE, SECTION III, APPENDIX G, FLAW SIZE

The critical flaw size in the bottom head region is typically larger than the critical flaw size in the beltline region because the bottom head, unlike the beltline, does not experience significant irradiation effects from exposure to neutron fluence. Although the thickness of the bottom head is smaller than the beltline region, the stresses due to internal pressure are curtailed in the bottom head because the smaller thickness is compensated by the spherical shape (as opposed to the cylindrical shape of the beltline). For the bottom head weld in question, there are no penetrations or appurtenances immediately adjacent to the weld that would necessitate the use of a stress concentration factor in determining the stresses. Thus, with lower stresses and a higher available toughness, the flaw tolerance capabilities of the bottom head region are expected to be significantly higher than the beltline region.

To demonstrate the increased flaw tolerance of the bottom head region, analyses were performed in accordance with ASME Code, Section ~~XI~~ III OK 3/11/96, Appendix G to establish curves of allowable pressure versus fluid temperature for various flaw sizes. These analyses utilized the limiting reference temperature for all plate and weld material in the bottom head region (RT_{NDT} = 16.9°F). Three flaw sizes were evaluated: 1/4T, 3/8T, and 1/2T. Various fluid temperature heatup and cooldown rates ranging from that identified on the current P-T Tech.

Spec. curve up to 100°F/hr were evaluated with respect to both outside surface and inside surface flaws. The limiting 100°F/hr results are shown in Figure 4 for all three flaw sizes, and are compared against the currently established Tech. Spec. P-T curve based on the limiting beltline region, for the hydrotest condition.

The results shown in Figure 4 demonstrate that the current P-T curve for the limiting beltline region significantly bounds that of the lower head region. For flaw sizes as large as 1/2T, significant margin remains. The primary reason for the significant difference between the bottom head and beltline regions is a result of decreased beltline toughness caused by irradiation. This effect is apparent when the adjusted reference temperature (ART) of 208°F (using the fluence specified on the Tech. Spec. P-T curve) for the limiting beltline material (upper shell plate) is compared to the 16.9°F value described above for the limiting bottom head material (head transition piece). Thus, the fracture toughness capability of the bottom head region is significantly bounded by the existing Tech. Spec. P-T curve for the beltline region.

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5. B&W-2108, Rev. 1, "Fluence Tracking System", B&W Nuclear Service Company, May 1992.

6. Procedure SP-296, "Documentation of Allowable Operating Transient Cycles", Rev. 11.

7. SI Calculation Package, FPC-01Q-301, "Reactor Pressure Vessel Finite Element Stress Analysis", Rev. 0.

8. ASME Section XI Task Group on Reactor Vessel Integrity Requirements, "White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions" EPRI TR-100251, January 1993.

9. ASME Boiler and Pressure Vessel Code, 1989 Edition, Section III, Appendices.

10. Design Report #6 "Thermal Mechanical Stress Analysis of Reactor Lower Head & Support Skirt", Vol. 2 of 2.

Table 1

Initial and Adjusted RT_{NDT} of Selected Plates and Welds at Crystal River Unit 3

Part Name & Material	MK No.	Heat No.	Slab No.	Estimated Initial RT_{NDT} °F	Chemistry Cu wt %	Chemistry Ni wt %	Chemistry Factor °F	Adjustments For 1/4 T: ΔRT_{NDT} °F	Margin σ_{Δ}, °F	Margin σ_{I}, °F	ART_{NDT} °F
Upper Shell SA-533-65 Grade B, Class 1 *(Limiting Beltline)*	A1-207-1	C4344	2	20	0.20	0.54	141.80	154.5	17.0	J 0	208.5
Head Transition Piece SA-508-64, Class 2 (Code Case 1332-3) *(Limiting Bottom Head)*	—	124W295VA1	—	10	0.10	0.80	67.00	3.5	1.7	0.0	**16.9**

	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4T	Fluence at ID	Attenuation, 1/4T $e^{-0.24x}$	Fluence 1/4 T	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Upper Shell	8.44	2.11	2.29E+19 n/cm^2	0.603	1.38E+19 n/cm^2	1.090
Head Transition Piece	5.00	1.25	4.79E+16 n/cm^2	0.741	3.55E+16 n/cm^2	0.052

Table 2

Stresses for Bottom Head-To-Transition Piece Weld

Distance, in	Pressure (2240 psi)		End of Heatup 100°F/hr			End of Cooldown 100°F/hr		
	Stress, ksi		Stress, ksi		Temp., °F	Stress, ksi		Temp., °F
	Axial	Hoop	Axial	Hoop	Note 1	Axial	Hoop	Note 2
0.00	19.670	15.080	0.152	-13.220	572.6	-2.536	13.730	301.1
0.50	19.500	15.030	0.423	-11.780	592.7	-2.227	12.530	306.3
1.00	19.290	15.000	0.573	-10.420	587.7	-1.803	11.390	311.0
1.50	19.080	14.970	0.575	-9.235	583.3	-1.277	10.390	315.1
2.00	18.860	14.930	0.405	-8.246	579.6	-0.616	9.573	318.7
2.50	18.650	14.900	0.056	-7.458	576.4	0.194	8.936	321.7
3.00	18.470	14.870	-0.460	-6.862	573.9	1.146	8.472	324.1
3.50	18.330	14.850	-1.132	-6.461	572.0	2.229	8.188	325.9
4.00	18.250	14.840	-1.936	-6.249	570.7	3.423	8.081	327.1
4.50	18.220	14.860	-2.857	-6.243	569.9	4.706	8.172	327.7
5.00	18.190	14.850	-3.804	-6.371	569.7	5.968	8.375	327.9

1. RCS Temperature at 604°F
2. RCS Temperature at 280°F

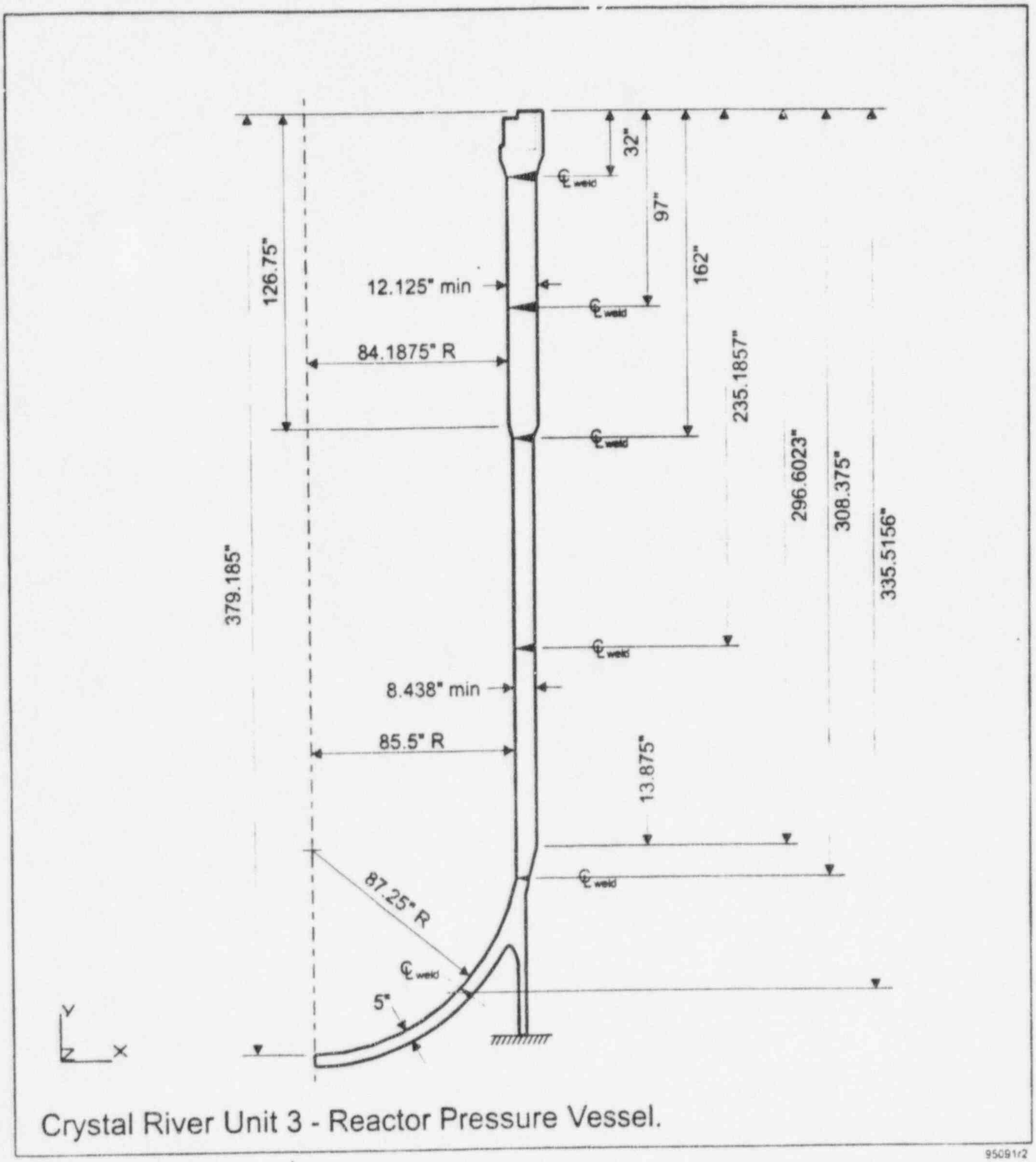

Figure 1. Vessel Geometry at CR-3 - Beltline and Bottom Head Regions

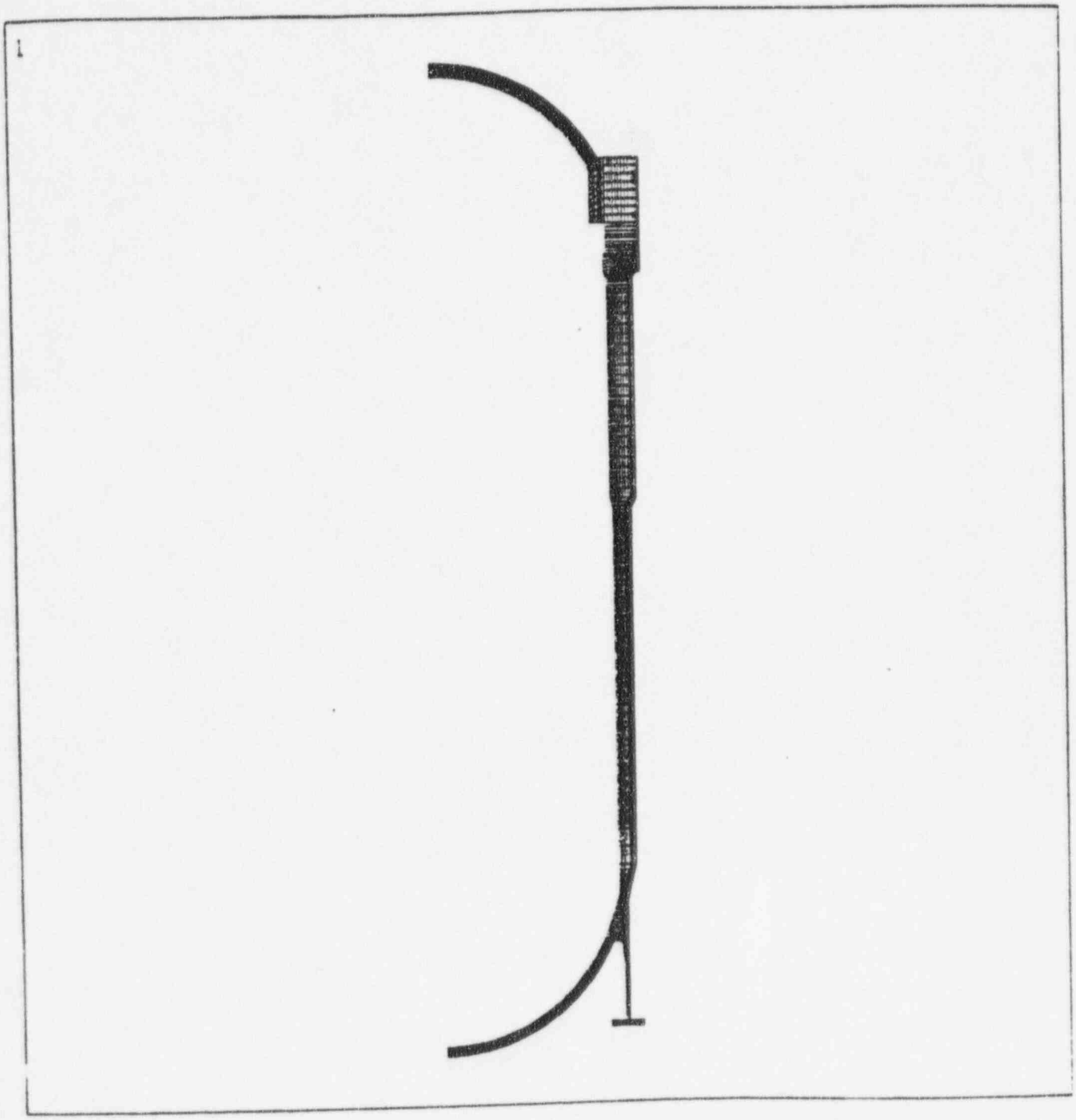

Figure 2. Reactor Vessel Axisymmetric Finite Element Model

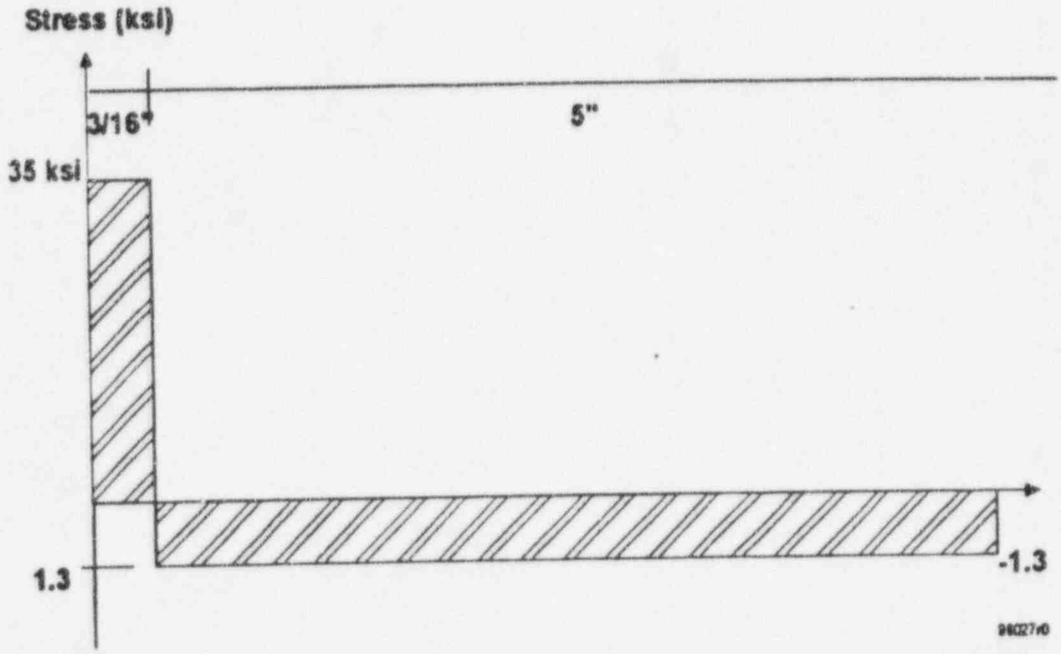

Figure 3. Clad Residual Stress

P-T Curve

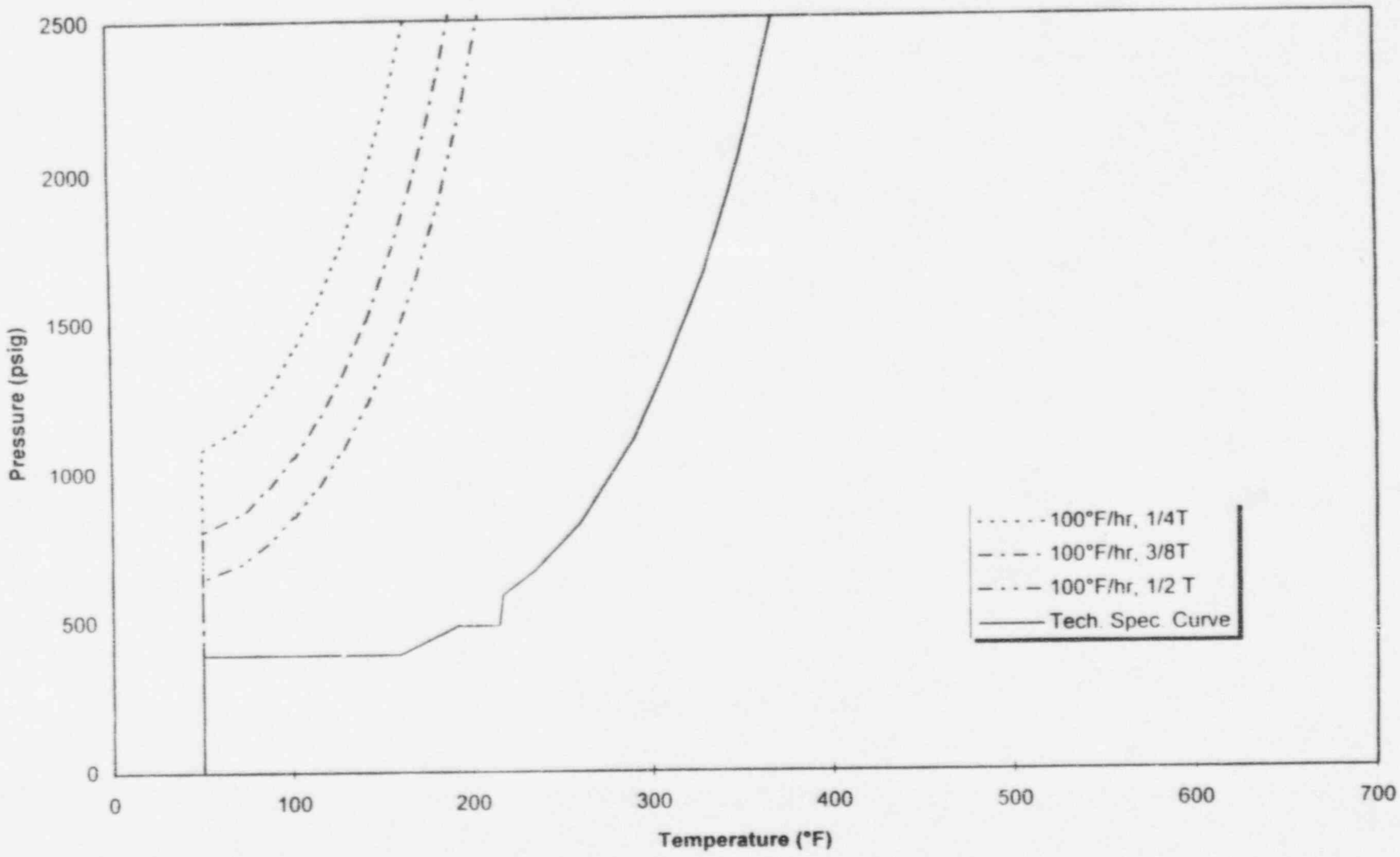

Figure 4. P-T Curve for Bottom Head Region for Flaw Sizes of 1/4T, 3/8T and 1/2T

P-T XLS [P-T Curve]

2/6/96