VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 3, 2001

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

Serial No. 01-637B
NL&OS/ETS R1
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION
ASME SECTION XI RELIEF REQUESTS SR-27, 28, 32 and 33
ALTERNATIVE REPAIR TECHNIQUE - REACTOR VESSEL HEAD

In a letter dated October 30, 2001 (Serial No. 01-637A), Virginia Electric and Power Company (Dominion) requested relief (Relief Requests SR-27 and SR-28 for Surry Unit 1 and SR-32 and SR-33 for Surry Unit 2) to use alternative repair techniques in the event that any flaws requiring repair in reactor vessel head penetrations were discovered during reactor vessel head penetration inspections. The inspections have been completed on the Surry Unit 1 and Unit 2 reactor vessel head penetrations. Six Unit 1 reactor vessel head penetrations had indications that required repair. These repairs have been completed. There were no Unit 2 reactor vessel head penetrations that required repair.

During a telephone conference call on November 6, 2001 with the NRC staff to discuss the subject relief requests, additional information was requested by the NRC to complete their review. Attachments 1 and 3 provide the proprietary version of the requested information. Attachments 2 and 4 provide the redacted version of the requested information. Attachment 5 provides the Affidavit for Withholding Proprietary Information from Public Disclosure.

Framatome ANP considers a portion of the requested information proprietary. In order to conform with the requirements of 10 CFR 2.790 concerning the protection of proprietary information, the information which is proprietary in the proprietary version is contained within brackets. Where the proprietary information has been deleted in the non-proprietary version, only the brackets remain (i.e., the information that was contained within the brackets in the proprietary version has been redacted.) The types of information Framatome ANP customarily holds in confidence is identified in Sections 6(a) through 6(e) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

A047

cc: U.S. Nuclear Regulatory Commission (Attachments 1 and 3 without enclosures)
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW
Suite 23 T85
Atlanta, Georgia 30303

Mr. R. A. Musser (Attachments 1 and 3 without enclosures)
NRC Senior Resident Inspector
Surry Power Station

Mr. R. Smith (Attachments 1 and 3 without enclosures)
Authorized Nuclear Inspector
Surry Power Station

Please contact Mr. Leslie Spain at (804) 273-2602 or Mr. Thomas Shaub at (804) 273-2763, if there are any questions about this submittal.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Commitments made in this letter:

Submit a procedure qualification record for welding P-No.3 Group No. 3 material to P-No. 43 material with F-No. 43 weld metal.

Attachments

1. Response to Request for Additional Information (Non-Proprietary) with enclosures:
   Weld Anomaly Considerations in the B&W CRDM ID Temper Bead Weld Repair (Non-Proprietary)
   Framatome ANP drawing 5015149 E (Proprietary)

2. Response to Request for Additional Information (Non-Proprietary) with enclosures:
   Weld Anomaly Considerations in the B&W CRDM ID Temper Bead Weld Repair (Non-Proprietary)
   Framatome ANP drawing 5015149 E (Redacted)

3. Summary of Structural Evaluation of Weld Repair of CRDM Housing (Non-Proprietary) with enclosures:
   Turkey Point CRDM Temperbead Bore Weld Analysis (Proprietary)
   Surry 1 and 2 Reconciliation with Turkey Point 3 RV Head and CRDM Nozzles (Non-Proprietary)
   Surry CRDMH Temperbead Weld Seismic Analysis (Non-Proprietary)
   Surry CRDM Nozzle IDTB Weld Anomaly Flaw Evaluation (Proprietary)
   Surry CRDM Nozzle 1.0" J-Groove Weld Flaw Evaluation (Proprietary)
   Surry CRDM J-Groove Weld Stress for Flaw Growth (1" Chamfer) (Proprietary)

4. Summary of Structural Evaluation of Weld Repair of CRDM Housing (Non-Proprietary) with enclosures:
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   Surry CRDM Nozzle 1.0" J-Groove Weld Flaw Evaluation (Redacted)
   Surry CRDM J-Groove Weld Stress for Flaw Growth (1" Chamfer) (Redacted)

5. Framatome ANP Affidavit for Withholding Proprietary Information from Public Disclosure

Attachment 1

Request for Additional Information

Relief Requests 27 and 32
Ambient Temperature Temperbead Weld Repair Technique (Non-Proprietary)

Relief Requests 28 and 33
Flaw Evaluation (Non-Proprietary)

With enclosures

Framatome ANP document 51-5012728-03, “Weld Anomaly Considerations in the B&W CRDM ID Temper Bead Weld Repair" (Non-Proprietary)

Framatome ANP document 5015149E, "Surry 1 and 2 CRDM Nozzle ID Temperbead Weld Repair" (Proprietary)

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Request for Additional Information
Relief Requests 27 and 32
Ambient Temperature Temperbead Weld Repair Technique

NRC Question

Provide a PQR for welding P-3 Group 3 metal to P-43 base metal with F-43 weld material.

Response

A procedure qualification for welding P-No. 3 Group No. 3 material to P-No. 43 material with F-No. 43 weld metal will be conducted and the results documented on a Procedure Qualification Record as soon as practical. Considering the limited resources presently available to conduct the procedure qualification due to the inspections and repairs ongoing at several nuclear units, it is anticipated that the PQR will be submitted to the NRC about March 1, 2002.

NRC Question

Relief Requests should mention if defects are detected in the weld repair by UT, a Section XI flaw evaluation will be performed for the weld repair with detected flaws.

Response

As a clarification to paragraph 4.0 (e) of the Relief Requests, any flaws detected by UT of the weld repairs will be evaluated in accordance with the requirements of ASME Section XI, IWB-3600.

NRC Question

NB 4622.11 states "whenever PWHT is impractical or impossible, limited weld repairs to dissimilar metal welds... may be made without PWHT..."

Provide a discussion (numerical comparison) of the radiation exposure differences between a Code required repair and the proposed alternative (Note: both Oconee 2 and TMI 1 have performed repairs using Framatome process on CRDMs with PWHT).

Response

The repair of 6 CRDM penetrations on the Surry Unit 1 vessel head using the machine ambient temperbead welding process incurred a total personnel exposure of 118 man-rem or about 20 man-rem per weld.

Because of the difficulty encountered in gaining access to the surface of the head due to the design of the insulation, it is estimated that removal of insulation, placement and removal of heating blankets, and conducting the necessary heating operations would add about 10% to 15% to personnel exposure. Experience at other plants, most notably Oconee, indicate that performing the repairs with purely manual techniques, which would involve preheat and post weld heating, could increase personnel exposure as much as another 50%.

NRC Question

Provide the following analyses:

Section III analysis of weld repair,

Section XI flaw evaluation for weld repair if flaw is detected by UT, and

Section XI flaw evaluation for remaining J-groove weld.

Response

A detailed proprietary summary of the structural evaluation of the weld repair for the CRDM housings is included in Attachment 2 of the letter. Attachment 3 provides the non-proprietary version of the evaluation.

Request for Additional Information

Relief Requests 28 and 33
Flaw Evaluation

NRC Question

Response to staff questions IWB 3142.4 and IWB 3420 was that it was impractical to perform volumetric characterization of the cracks in the J-groove weld. And there was some discussion on contouring the corner of the J-groove weld.

Provide a description of the contouring including expected cross-sectional area removal. Discuss characterization and recording of cracks revealed by PT exam of the J-groove weld area and of the J-groove weld machined surface.

Response

Upon completion of the repair weld, the remaining J-groove weld which was not removed by the machining operation that cut away the lower piece of the penetration will be chamfered by hand. This activity removes a substantial portion of the remaining weld. The total amount of material to be removed depends upon which penetration is repaired and also on the location on the penetration since the contour of the J-groove weld varies around the penetration because of the oblique angle most of them make with the head. Pertinent details are shown on Framatome ANP drawing 5015149 E, enclosed.

After the lower portion of the penetration tube is machined away and prior to repair welding, the area from 1/2 inch above the repaired weld to the bottom of the remnant J-groove weld will be liquid penetrant inspected. Any indications noted in the remnant weld will be recorded. Subsequent to the chamfering operation of the remnant weld, it will be assumed that a corner flaw exists equal in depth to the original J-groove weld width minus the removed material (about 1.053 inches in the worst case). Analysis of this assumed flaw is described in Attachment.

NRC Question

Explain the effect of the anomalies at the triple point (carbon steel vessel, Inconel 600 CRDM, and Inconel 690 weld material) on NDE. Describe the type of defect, if any, found at these anomalies.

Response

Please see enclosed Framatome ANP document 51-5012728-03, "Weld Anomaly Considerations in the B&W CRDM ID Temper Bead Weld Repair," for a discussion of the kind and size of flaw anticipated at the triple point. There are no anticipated effects specific to the anomaly at the triple point. The anomaly acts as any other lack of fusion indication type reflector and has been shown to be readily detectable with the UT techniques, which will be employed.

NRC Question

Enclosure 1 1.0(e) references Cases used in the repair/replacement plan. Does the word “cases” mean Code cases endorsed or authorized by the NRC.

Response

Yes, but for these relief requests, none of the Code Cases previously approved for the Surry units apply.

Attachment 1 enclosure

Framatome ANP document 51-5012728-03, "Weld Anomaly Considerations in the CRDM ID Temper Bead Weld Repair" (Non-Proprietary)

FRAMATOME ANP

ENGINEERING INFORMATION RECORD

Document Identifier 51 - 5012728 – 03

Title Weld Anomaly Considerations in the CRDM ID Temper Bead Weld Repair

PREPARED BY:		REVIEWED BY:	
Name	J.R. Dorman, Jr.	Name	H. W. Behnke
Signature	JR Dorman Jr. Date 10/28/01	Signature	H Behnke Date 10/29/01

Technical Manager Statement: Initials MSC

Reviewer is Independent.

Remarks:

This document provides a description and the assumptions associated with the presumed existence of a weld anomaly in the CRDM nozzle ID Temper Bead Weld Repair. The description and assumptions will be considered for service suitability in a separate F-ANP calculation package.

Initial Issue

Revision 1 – Minor editorial changes. Additional description of anomaly including void considerations.
Revision 2- Editorial revisions to delete direct reference to associated calculation packages. This document was an input to the packages and not associated with the calculation results.
Revision 3- Removed B&W from title and text since the EIR is pertinent to all FRA-ANP ID temper bead repairs performed on CRDMs.

An artifact of the 360-degree temper bead weld repairs of Control Rod Drive Mechanism nozzles is an anomaly in the weld at the triple point. The triple point is the juncture of the low alloy head, the alloy 600 nozzle and possibly a previously applied alloy 52 weld bead. The previous weld bead possibility arises from the variation in depth of the machined bore. The first weld layer is on the low alloy steel bore and may not actually weld on the triple point. The second layer would then be the tie-in at the triple point. Due to a combination of factors, this area of the weld has crack-like indications that could be 360 degrees around the nozzle. The crack-like indication extends from the existing crevice into the weld at angles from 0 to 90 degrees, where 90 degrees is in the through-thickness direction of the nozzle and zero degrees is along the low alloy fusion line. Mock-up testing has verified that the anomalies are common and do not exceed .1 inch in length. The typical length is closer to .05 inches. The anomaly may consist of a void with a crack-like indication extending from the void. The combined indication is still less than .1 inch. The void is principally a lack of fusion to the low alloy steel at the triple point. The anomaly is a crack-like indication extending into the weld and is not a lack of fusion to the nozzle wall. It is assumed for conservatism that a .1 inch indication could exist in the through-thickness direction of the nozzle. Indications of this conservatively large size have not been observed. Due to its crack-like configuration, this indication is analytically treated as a flaw. The Section III analysis is not affected and consideration of the flaw is made in the flaw evaluation.

An evaluation will be prepared to justify the as-left condition including the indications up to .1 inch. The evaluation also includes a hypothetical planar flaw normal to the circumferential stress. This flaw will be assumed to be a 2-to-1 elliptical flaw with the minor axis equal to .1 inch. This type of flaw is not expected but is included for completeness since the circumferential stresses are the largest stresses present in the weld area. This evaluation will be prepared in accordance with ASME Section XI and will demonstrate that for the intended service life of the repair, the fatigue crack growth is acceptable and the crack-like indications remain stable. These two findings will satisfy the Section XI criteria but will not include considerations of stress corrosion cracking such as primary water stress corrosion cracking (PWSCC) or residual stresses.

Since the crack-like defects are not exposed to the primary coolant and the air environment is benign for the materials at the triple point, the time-dependent crack growth rates from PWSCC are not applicable regardless of residual stresses.

Residual stresses may also require consideration for ductile tearing when operating stresses are superimposed. The residual stress field by itself cannot promote ductile tearing or it would not be stable during welding. The anomalies have been shown to be stable by welding mock-ups simulating the actual geometry and materials. Even though the residual stresses for this type of weld would be very complex, it is apparent that by the size of the weld and the nature of the restraint that the residual stresses would have limited effect on driving a crack. The weld residual stresses are not like piping thermal expansion stresses where there may be considerable stored energy in long runs of pipe. The weld residual stresses are imposed by the inability of the weld bead to shrink to a nominal strain condition upon cooling. The attachment of the weld to the surrounding material generally promotes tensile stresses in the bead upon cooling. However even though the stresses are generally at the yield strength, the accompanying strains are not large due to the limited size of the beads and in this case the total size of the weld.

It is concluded that the residual stress field will produce minimal ductile tearing The NiCrFe materials are extremely crack-tolerant when not in an aggressive environment and the ASME Section XI evaluation performed for fatigue growth and net section failure will be adequate. Residual stresses need not be considered because PWSCC effects are not applicable, and the geometry is not conducive to sustained ductile tearing.

Attachment 2

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FRAMATOME ANP

ENGINEERING INFORMATION RECORD

Document Identifier 51 - 5012728 – 03

Title Weld Anomaly Considerations in the CRDM ID Temper Bead Weld Repair

PREPARED BY:

Name J.R. Dorman, Jr.

Signature JRDorman Jr Date 10/28/01

REVIEWED BY:

Name H. W. Behnke

Signature HBehnke Date 10/29/01

Technical Manager Statement: Initials MSC

Reviewer is Independent.

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Attachment 2 enclosure

Framatome ANP document 5015149E, "Surry 1 and 2 CRDM Nozzle ID Temperbead Weld Repair" (Redacted)

REV	DESCRIPTION	DATE	APPROVAL
1	REVISED DIMENSIONS G, H, J, AND K IN TABLE 1; (.750) REMAINING WELD LENGTH IN STEP 6 WAS (.500).	11/19/01	WSM DEM MAS
2	REVISED DIMENSIONS H AND K IN TABLE 1; (1.000) REMAINING WELD LENGTH IN STEP 6 WAS (.750).	11/24/01	WSM DEM MAS

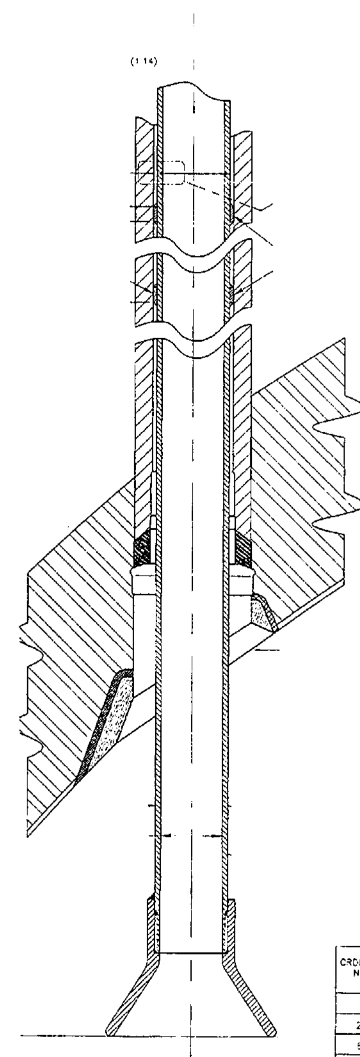

TABLE 1

CRDM NOZZLE NUMBER	A	B	E	F	G	H	J	K	R	Y	Z
1	9.013	38.000	8.513	6.663	.650	.658	.580	.558	0.000	27.982	33.487
2 THRU 5	9.226	38.906	8.526	6.972	.571	.682	.570	.669	11.973	27.076	32.987
6 THRU 9	9.489	39.844	8.989	7.130	.575	.714	.572	.679	16.932	26.138	32.095
10 THRU 13	9.717	41.719	9.217	7.347	.563	.764	.560	.712	23.945	24.263	30.419
14 THRU 21	9.831	42.688	9.331	7.456	.559	.790	.556	.730	26.772	23.293	29.545
22 THRU 25	10.077	45.564	9.577	7.684	.562	.872	.540	.790	33.864	20.389	26.827
26 THRU 29	10.157	46.656	9.657	7.924	.550	.901	.542	.811	35.918	19.325	25.868
30 THRU 37	10.163	47.656	9.663	7.886	.547	.932	.542	.834	37.901	18.325	24.935
38 THRU 45	10.303	50.844	9.803	8.167	.559	1.033	.538	.911	43.168	5.138	21.979
46 THRU 49	10.512	54.156	10.012	8.428	.563	1.145	.534	.998	47.691	11.825	19.865
51, 53, 55, 57	10.624	55.313	10.124	8.531	.571	1.192	.534	1.030	49.366	10.668	17.790
58 THRU 61	10.753	56.300	10.253	8.648	.578	1.237	.534	1.054	50.796	9.492	15.697
62 THRU 69	10.978	58.906	10.478	8.652	.564	1.340	.502	1.135	53.544	7.075	14.443

DWN BY J G SPARKS
CHK'D BY J G SPARKS
PASSED BY C L MATTHEWS
APPV'D BY M S CARPENTER
DATE 10/21/01
CONTRACT NO 1231180

SURRY 1 & 2
CRDM NOZZLE
ID TEMPER BEAD
WELD REPAIR

THIS DRAWING IS THE PROPERTY OF FRAMATOME ANP

SCALE NONE
DRAWING NO 5015149 E
REV NO 2

22162 (01/01)

Attachment 3

Summary of Structural Evaluation of Weld Repair of CRDM Housings
(Non-Proprietary)
with the following enclosures:

Turkey Point CRDM Temperbead Bore Weld Analysis (Proprietary)
Surry 1 and 2 Reconciliation with Turkey Point 3 RV Head and CRM Nozzles (Non-Proprietary)
Surry CRDMH Temperbead Weld Seismic Analysis (Non-Proprietary)
Surry CRDM Nozzle IDTB weld Anomaly flaw Evaluation (Proprietary)
Surry CRDMNozzle 1.0" J-Groove Weld Flaw Evaluation (Proprietary)
Surry CRDM J-Groove Weld Stress For Flaw Growth (1" Chamfer) (Proprietary)

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

SUMMARY OF STRUCTURAL EVALUATION OF WELD REPAIR OF CRDM HOUSINGS SURRY POWER STATION UNIT 1

0.1 OBJECTIVE:

The objective of this summary is to document the review of the structural evaluation of the repair of the following six CRDM housings on the reactor head of Surry Power Station Unit 1: S-1-18, S-1-27, S-1-40, S-1-47, S-1-65, S-1-69.

0.2 INTRODUCTION AND BACKGROUND:

Due to the recent experience of degradation of CRDM nozzle housing in the vicinity of the J-groove weld to the reactor vessel head described in NRC Bulletin 2001-01, Dominion has inspected the CRDM housing penetrations to the reactor head for Surry Unit 1. The inspection revealed evidence of degradation at the J-Groove weld and possible leakage at the six CRDM housing penetrations cited above. Framatome ANP was contracted by Dominion to repair the nozzles.

Repair has been performed to meet the applicable configuration requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NB, 1989 edition. The repair weld has been deposited using the machine GTAW process with cold wire feed, in accordance with the ASME Section XI, IWA-4000 with modification as described in by Relief Requests SR-27 and SR-28.

The repair effort followed several steps, not necessarily in the order given below. A baseline volumetric and surface examination was performed for the repair region. The lower portion of the thermal sleeve was cut and removed with automatic tools after cleaning. The CRDM nozzle was rolled into the reactor vessel head penetration. The lower end of the nozzle was machined away into the head to make the weld preparation beyond the degraded area. The J-weld at the bottom end of the penetration was chamfered by grinding to remove part of the degraded weld. The bored region of the head and weld prep on the bottom of the remaining portion of the CRDM nozzle were examined by PT. The repair area was cleaned for welding and weld material was deposited. The repair weld was machined to reestablish a nozzle free path and to provide a suitable surface for PT and UT. PT and UT examinations were performed for the repair. The repair was remediated using an abrasive water-jet. The thermal sleeve was replaced as the last step of the repair.

The portion of the reactor vessel head (RVH) containing the CRDM nozzle is fabricated from SA-533 Grade B, Class 1. The portion of the CRDM nozzle that penetrates the RVH is SB-167 Alloy 600. The weld material for the repair is ERNiCrFe-7, UNS N06052. The cobalt content of the weld filler material was limited to 0.2%. The replacement thermal sleeve has been welded to the upper sleeve using metal insert in accordance with SFA 5.9 ER309L or ER316L per ASME Section II.

Three different structural evaluations have been performed to establish the structural integrity of the repair and design life of the repair:

1) Stress analysis of the repair has been performed conforming to the requirements

of ASME, Section III, Subsection NB, Paragraph NB-3000, 1989 Edition.

2) A fracture mechanics analysis has been performed in accordance with IWB-3132.4 and IWB-3600 of ASME Section XI Code. This analysis considered a 0.100-inch weld anomaly and assumed it to be a linear defect and extending into the repair weld in any direction at the triple point. The triple point is defined as the intersection of the reactor head base material, the CRDM nozzle, and the repair weld. It has been justified by experience that the assumed flaw is bounding.

3) A fracture mechanics analysis has also been performed to justify a postulated flaw remaining in the J-groove weld remnant between the original CRDM nozzle and the reactor vessel head. This analysis is important because the flaw in the remaining weld cannot be characterized by available NDE methods. The size of the flaw considered in this analysis is equal to the largest radial length through the remaining J-weld. The flaw growth analysis has been used as one of the considerations to establish design life of the repair.

These three analyses are summarized below. The summary includes the configurations analyzed, loading conditions, design criteria, and code compliance. The details of stresses, cumulative usage factors, flaw tolerance and flaw growth analyses are presented. Based upon the results of these conservative analyses, the design life of the repair is predicted to be at least five years. The life of the repair is dependent on the size of the remaining J-groove weld, where the analysis conservatively postulated an initial flaw through the remaining thickness of the weld.

1. ASME SECTION III ANALYSIS OF REPAIR

1.0 OBJECTIVE

The purpose of this review is to summarize the ASME Section III analyses that have been performed for the CRDM temperbead bore weld repair for Surry Unit 1 Reactor Vessel Upper Head Penetrations S-1-18, -27, -40, -47, -65, and -69. The repair consists of cutting the CRDM housing above the original attachment weld, removing the lower portion of the housing and welding the remaining housing to the reactor vessel upper head with a temperbead weld. Analyses have been performed that demonstrate that the repair design meets the applicable requirements of the ASME Code Section III. The Surry CRDM nozzles are similar to corresponding nozzles analyzed previously for this repair procedure. A formal reconciliation was performed to allow use of these previous analyses for Surry.

1.1 GEOMETRY/FINITE ELEMENT MODEL DEFINITION

The finite element model used to analyze the CRDM housing nozzle to reactor vessel upper head weld region is documented in References 1-1, 1-2, and 1-3. The finite element model is a 3-dimensional model of a 180-degree segment of a CRDM tube with the adjacent head region and interconnecting weld.

1.2 MATERIALS

The materials of the components in the finite element model are summarized below (References 1-1, 1-2, and 1-3):

Reactor Vessel Head Base Metal = ASTM A533, Grade B, Class 1 (Mn-Mo Steel)
CRDM Housing Nozzle = ASME SB-167 Inconel
Cladding = Stainless Steel
J-Groove Buttering = Alloy 600 (Inconel)
J-Groove Filler = Alloy 600 (Inconel)
Repair Weld = ERNiCrFe-7, UNS N06052 Per ASME Section II, Part C, SFA-5.14, with properties similar to Alloy 690.

1.3 LOADS

The loads considered in the design of the CRDM IDTB (ID Temperbead) weld repair are based on those considered in the original design specification (Reference 1-5) and design report (Reference 1-6) for the reactor vessel top head and CRDM housings. The loads considered are:

Design Pressure/Temperature
Plant heatup and cooldown at 100ºF/hr.
Plant loading and unloading at 5% of full power per minute
Small step load increase and decrease
Large step load decrease
Loss of load
Loss of power
Loss of Flow
Reactor Trip from full power
Turbine roll test
Primary side hydrostatic test at 3105 psig
Primary side hydrostatic test at 2485 psig
Steady state fluctuations
Steam pipe break (faulted)
OBE seismic loading
DBE seismic loading

For analysis purposes, operational transients have been grouped into three separate analyses: 1) heatup/cooldown, 2) plant loading/unloading, and 3) remaining (or rapid) transients. For the plant loading/unloading transient, the ASME Section III fatigue evaluation for the IDTB weld repair has assumed a total of 14,500 loading/unloading events over the plant design life. While this assumption does not bound the 29,000 cycles assumed in the original design specification, it is bounding relative to actual plant operation. The 29,000 cycles of loading and unloading was based on load-following operation. Surry has operated (and will continue to operate) in a base-load capacity manner, which results in significantly fewer loading/unloading cycles. The assumed value of 14,500 cycles is still very conservative. The rapid transient has been defined to bound the small step increase/decrease, large step load decrease, loss of load, loss of power, loss of flow, and reactor trip operational transients. The transients used in the analyses have been reviewed and determined to envelop the design transients for Surry.

1.4 LOADING CONDITIONS/ STRESS CRITERIA:

The following loading conditions and stress criteria are used in the evaluation documented in Reference 1-3. The 1989 Edition of the ASME Code (No Addenda), Section III (Reference 1-4) is used for the evaluation.

Primary Stress Intensities for Design Conditions:

NB-3221.1, Primary General Membrane Stress Intensity ($P_m \leq S_m$)

NB-3221.2, Local Membrane Stress Intensity ($P_l \leq 1.5\ S_m$)

NB-3221.3, Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 1.5\ S_m$)

Primary + Secondary Stress Intensity Range for Service Level A/B (normal/upset) Conditions:

NB-3222.2, Primary + Secondary Stress Intensity Range (P + S Stress Intensity Range $\leq 3\ S_m$)

Fatigue Usage

NB-3222.4, Fatigue Usage ≤ 1.0

Primary Stress Intensities for Emergency (Level C) Conditions:

NB-3224.1, Primary General Membrane Stress Intensity ($P_m \leq 1.2\ S_m$)

NB-3224.1, Local Membrane Stress Intensity ($P_l \leq 1.8\ S_m$)

NB-3224.1, Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 1.8\ S_m$)

Primary Stress Intensities for Faulted (Level D) Conditions:

NB-3225, F-1331.1(a), Primary General Membrane Stress Intensity ($P_m \leq 0.7\ S_u$)

NB-3225, F-1331.1(b), Local Membrane Stress Intensity ($P_l \leq 1.05\ S_u$)

NB-3225, F-1331.1(c), Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 1.05\ S_u$)

Primary Stress Intensities for Test Conditions:

NB-3226(a), Primary General Membrane Stress Intensity ($P_m \leq 0.9\ S_y$)

NB-3226(b), Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 2.15\ S_y - 1.2P_m$)

The repair is analyzed to 1989 version of ASME Section III Code (Reference 1-4). The original stress report (Reference 1-6) was prepared conforming to the requirements of 1968 version of ASME Section III Code (Reference 1-8). The stress criteria of the original design differ from the 1989 version of Section III Code only for allowable stresses in OBE and SSE conditions. In the original design, the stress in the OBE condition was checked against an allowable stress intensity of 1.2 S_m and SSE condition was checked against an allowable stress intensity of 1.8 S_m. In order to comply with the original design criteria, the stresses under seismic loading (Reference 1-7) were also compared with the original Code allowable.

1.5 RESULTS:

The results of the ASME Section III analysis of the weld repair are summarized below:

Primary Stress Intensities for Design Conditions (Design Pressure at Design Temperature):

RV Head: $P_m = 16.6$ ksi $\leq S_m = 26.7$ ksi
$P_l = 20.4$ ksi $\leq 1.5\ S_m = 40.1$ ksi
$P_l + P_b = 25.6$ ksi $\leq 1.5\ S_m = 40.1$ ksi

Nozzle/Weld: $P_m = 6.2$ ksi $\leq S_m = 23.3$ ksi
$P_l = 9.85 \leq 1.5\ S_m = 35.0$ ksi
$P_l + P_b = 9.85$ ksi $\leq 1.5\ S_m = 35.0$ ksi
(Also less than 1.2 S_m=27.96 ksi)

Normal/Upset Service Level (A/B) Condition

Primary + Secondary Stress Intensity Range:
Heatup/Cooldown Transient: $S_n = 36.7$ ksi $\leq 3\ S_m = 80.0$ ksi
Loading/Unloading Transient: $S_n = 16.1$ ksi $\leq 3\ S_m = 80.0$ ksi
Rapid (Remaining) Transient: $S_n = 9.1$ ksi $\leq 3\ S_m = 80.0$ ksi

Fatigue Usage

The total fatigue usage, based on an assumed fatigue strength reduction factor of 4.0, for a 14-year service life is calculated to be 0.525. With this result, the qualified operating life for which the fatigue usage is less than 1.0 is 26.7 years.

Emergency (Level C) Conditions:

RV Head:
Maximum Allowable Pressure Based on P_m Limit = 4,819 psi
Maximum Allowable Pressure Based on P_l Limit = 5,895 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 4,697 psi

Nozzle/Weld:
Maximum Allowable Pressure Based on P_m Limit = 11,089 psi
Maximum Allowable Pressure Based on P_l Limit = 16,633 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 16,633 psi

All of the maximum allowable pressures based on the Emergency (Level C) condition stress limits are greater than the maximum hydrotest pressure of 3105 psi. The level C pressure loading is not specified for Surry.

Faulted (Level D) Conditions:

RV Head:
Maximum Allowable Pressure Based on P_m Limit = 8434 psi
Maximum Allowable Pressure Based on P_l Limit = 10,294 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 8,203 psi

Nozzle/Weld:
Maximum Allowable Pressure Based on P_m Limit = 22,540 psi
Maximum Allowable Pressure Based on P_l Limit = 33,830 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 33,830 psi

All of the maximum allowable pressures based on the Faulted (Level D) condition stress limits are greater than the maximum hydrotest pressure of 3,105 psi. The level D pressure loading is not specified for Surry.

Primary Stress Intensities in SSE Condition

RV Head:
Insignificant Seismic effect

Nozzle/Weld:
$P_l + P_b = 15.45$ ksi $\leq 2.4\ S_m = 55.9$ ksi.
(Also $\leq 1.8\ S_m = 41.94$ ksi)

Test Conditions:

RV Head:
Maximum Allowable Pressure Based on P_m Limit = 6,777 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 5,225 psi

Nozzle/Weld:

Maximum Allowable Pressure Based on P_m Limit = 12,702 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 15,121 psi

All of the maximum allowable pressures based on the test condition stress limits are greater than the maximum hydrotest pressure of 3,105 psi. No hydrotest of this level is planned for Surry.

1.6 CONCLUSION:

The CRDM housing nozzle temperbead weld repair design meets the stress and fatigue requirements of the ASME Code, Section III, 1989 edition w/o Addenda. The conservative fatigue analysis indicates that the repair design has a qualified operating life of at least 26.7 years.

1.7 REFERENCES:

1-1 Framatome ANP Document No. 32-5014129-00, "Turkey Point - CRDMH 3D FE Model."
1-2 Framatome ANP Document No. 51-5015197-01, "Surry 1 & 2 Reconciliation with Turkey Point 3 RV HD & CRM Noz." (Included as Enclosure 1-2)
1-3 Framatome ANP Document No. 32-5014640-00, "Turkey Point – CRDM Temperbead Bore Weld Analysis." (Included as Enclosure 1-1)
1-4 ASME Boiler and Pressure Vessel Code, 1989 Edition, Section III, No Addenda
1-5 Surry Reactor Vessel Design Specification 676499, Rev. 1, "Addendum to Equipment Specification 676413, Rev. 1, Surry Power Station 1."
1-6 Calculation 30678-1130, "Reactor Vessel – Final Stress Report (Parts I & II), Surry Power Station Units 1 and 2, " Rotterdam Dockyard Company.
1-7 Framatome ANP Document No. 32-5015624-00, "Surry CRDMH Temperbead Weld Seismic Analysis." (Included as Enclosure 1-3)
1-8 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1968 Edition to and including Winter 1968 Addenda

2. SURRY CRDM NOZZLE IDTB WELD ANOMALY FLAW EVALUATIONS

2.1 PURPOSE:

This review summarizes the CRDM nozzle IDTB weld anomaly flaw evaluation. This is a common evaluation for IDTB weld repair performed on the following six CRDM nozzles of Surry Power Station Unit-1: S-1-18, S-1-27, S-1-40, S-1-47, S-1-65, and S-1-69.

2.2 CONFIGURATION:

A fracture mechanics evaluation has been performed for a postulated weld anomaly in the CRDM nozzle IDTB weld repair design (Reference 2-1). During the welding process a maximum of 0.1" weld anomaly may be formed due to lack of fusion at the triple point.

The postulated weld anomaly is a 0.1" semi-circular region of lack of fusion extending 360-degrees around the circumference at the triple point location at the intersection of three materials: the Alloy 600 nozzle, the Alloy 52 weld, and alloy steel head. The flaw evaluation simulated the defect as a 360-degree circumferential crack of depth of 0.1" on the OD of a circular tube. The evaluation also postulated an axially oriented semi-circular OD surface flaw with depth equal to 0.1" and axial length of the flaw equal to 0.2". Both of these circumferential and axial flaws postulated on the outer surface propagate horizontally into the weld material. A semi-circular, cylindrically oriented flaw is also postulated along the interface between the weld and head, and propagates downward between the two components. The finished thickness of the wall used in the analysis is 0.488".

2.3 MATERIAL PROPERTIES:

Fracture toughness curves for SA-533 Grade B, Class 1 material are illustrated in the ASME Section XI, Code, 1989 in Figure A-4200-1. At an operating temperature of 600°F, the K_{Ia} fracture toughness value for this material is above 200 ksi√in for assumed RT_{NDT} of 60°F. The toughness properties of Alloy 600 and weld material are better than 200 ksi√in and; therefore, an upper-shelf value of 200 ksi√in is used in the analysis (Reference 2-1).

2.4 LOADS:

The transient loads applicable for evaluation of this repair were conservatively grouped into three categories:

Heatup/Cooldown	3.33 cycles per year
Plant Loading/Unloading	250 cycles/year
Remaining rapid transients	46.67 cycles per year

2.5 APPLICABLE CRITERIA:

The flaw acceptance is based on the 1989 ASME Code Section XI criteria for applied stress intensity factor (IWB-3612) and limit load (IWB-3642). For flaw growth analysis in the RV Head, Article A-4300 of Section XI code is used. For flaw growth rate in the repair weld Article C-3210 of Section XI (normally applicable to austenitic stainless steel in an air environment) has been used.

2.6 RESULTS:

The results of the analyses showed:
A minimum fracture toughness margin of 11.4 compared to the required margin of √10 per IWB-36-12.

A margin on limit load of 6.25, compared to the required margin of 3.0 per IWB-3642.

Fatigue crack growth is minimal. The predicted crack growth over 25 years is from 0.100" to 0.114". There is no acceptance standard for this. However, the predicted crack will still remain shallow. (Details of evaluation are provided in Enclosure 2-1.)

2.7 CONCLUSION:

The IDTB weld repair will maintain structural integrity for the predicted life of repair.

2.8 REFERENCE:

2-1 Framatome ANP, Document No. 32-5015219-00, "SURRY CRDM NOZZLE IDTB WELD ANOMALY FLAW EVALUATIONS." (Included as Enclosure 2-1)

3. FLAW EVALUATION OF THE REMAINING J-GROOVE WELD

3.1 OBJECTIVE:

The purpose of this review is to summarize the flaw evaluation of the remaining J-groove weld following the IDTB weld repair of the following six CRDM nozzles of Surry Power Station Unit-1: S-1-18, S-1-27, S-1-40, S-1-47, S-1-65, and S-1-69.

3.2 BACKGROUND:

Since a potential flaw in the J-groove weld cannot be sized by currently available NDE techniques, it must be assumed that the as-left condition of the remaining J-groove weld includes degraded or cracked weld material extending through the entire J-groove weld and Alloy 182 butter material.

The hoop stresses in the J-groove weld are generally about twice the axial stress; therefore, the preferential direction for cracking is radial out from the bore radius. It is postulated that a radial crack in the Alloy 182 weld metal would propagate through the weld and butter, to the interface with the low-alloy steel head. Extensive industry experience has shown that flaws originating in an Alloy 82/182 weld have not propagated into the ferritic base material, and it is fully expected that such a crack would then blunt and arrest at the butter-to-head interface. However, for this evaluation, it is conservatively assumed that the stress corrosion crack in the weld would combine with a small flaw in the reactor head steel to form a large radial corner flaw that would propagate into the low alloy head by fatigue crack growth under cyclic loading conditions.

3.3 CONFIGURATION:

Analytically, this flaw has been simulated using a corner flaw model (Reference 3-1). The repair incorporates a chamfer at the inside corner of the remnant J-groove weld to limit the potential crack length through the weld from the inside corner of the bore chamfer to the low alloy steel vessel head. The evaluation assumes the initial flaw depth as 1.053 inch, which represents the distance completely through the remaining weld.

3.4 MATERIAL PROPERTIES:

Fracture toughness curves for SA-533 Grade B, Class 1 material are illustrated in the ASME Section XI, Code, 1989 in Figure A-4200-1. At an operating temperature of 600°F, the K_{Ia} fracture toughness value for this material is above 200 ksi√in for assumed RT_{NDT} of 60°F. The toughness properties of Alloy 600 and weld material are better than 200 ksi√in and; therefore, an upper-shelf value of 200 ksi√in is used in the analysis.

3.5 APPLICABLE CRITERIA:

The flaw acceptance is based on the 1989 ASME Code Section XI criteria for applied stress intensity factor (IWB-3612).

3.6 LOADINGS:

The imposed stress distribution was obtained from a 3-D ANSYS finite element analysis, which was performed to determine operating transient stresses in the vicinity of the CRDM nozzle following the repair (Reference 3-2). Previous analyses had determined that the outermost nozzles with the largest "hillside angle" (the relative angle between the local plane of the reactor head and the nozzle vertical centerline) experience the greatest increase in stress in the region of the J-groove weld. Therefore, the finite element model represented one of the outermost nozzles, and the results will conservatively bound all nozzle locations that have a smaller hillside angle. The finite element analysis found that the highest stresses occur at the uphill side of the nozzle along the vertical plane formed by the centerlines of the nozzle and the reactor. Transient analyses were performed for normal heatup and cooldown cycles, plant loading and unloading cycles, reactor trip, and other rapid transients. The maximum stresses were determined along a line into the reactor head material from the uphill "corner" of the nozzle bore, representing the progression of the crack front of the assumed corner crack.

Residual stresses were not explicitly included in this flaw evaluation, since a crack that has propagated all the way through the weld would tend to relieve these stresses, and a crack at the butter-to-head interface would experience only compressive residual stress ahead of the crack.

The fracture mechanics analysis was performed assuming the following pattern for accumulating cycles:

Transient	Frequency (cycles / year)
Heat up / Cool down	3.33
Plant Loading / Unloading	50.00*
Large Step Decrease	3.33
Loss of Load	1.33
Loss of Flow	1.33
Reactor Trip	6.67
Remaining Transients	34.00

* The original design specification included 29,000 cycles of plant loading/unloading for the life of the plant. As discussed previously, the number of cycles in the design specification was conservatively based on load-following operation. The 50 cycles/year is conservative for the actual base load capacity mode of operation under which Surry has operated and will continue to operate.

3.7 RESULTS:

The crack growth analysis was performed for each set of transients for each year and iteratively summed by linking the incremental crack growth for each of the sets of transients for each year. The results are compared to the fracture toughness requirements of Section XI. Applying the conservatively assumed number of cycles per year, the fracture mechanics analysis shows that the crack will be acceptable for over five years of operation. The flaw depth at the end of five years is projected to be 1.123". The calculated stress intensity factor at the final flaw size for the most severe transient is less than K_I = 63.16 ksi -√ in, compared to the fracture toughness upper-shelf value of K_{Ia} = 200.0 ksi -√ in. This provides a safety margin of 3.17, which is greater than √10 safety margin required by Article IWB-3612 of the Code.

(Details of the fracture mechanics analysis are given in Enclosure 3-1. Information on the stress analysis is provided in Enclosure 3-2.)

3.8 REFERENCES:

3-1 Framatome ANP Document No. 32-5015650-00, "SURRY CRDM NOZZLE 1.0" J-GROOVE WELD FLAW EVALUATION." (Included as Enclosure 3-1)

3-2 Framatome ANP Document No. 32-5015651-00, " SURRY-CRDMH J-GROOVE WELD STRESS FOR FLAW GROWTH," (1" CHAMFER), (Included as Enclosure 3-2)

Enclosure 1-2
(Non-Proprietary)

Framatome ANP Document No. 51-5015197-01,
"Surry 1 & 2 Reconciliation with Turkey Point 3 RV Hd & CRM Noz."

FRAMATOME ANP

ENGINEERING INFORMATION RECORD

Document Identifier 51 - 5015197 - 01

Title SURRY 1 & 2 RECONCILIATION WITH TURKEY POINT 3 RV HD & CRM NOZ.

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Technical Manager Statement: Initials ADM

Reviewer is Independent.

Remarks:

Purpose:

This report documents the applicability of engineering analyses performed for the Turkey Point 3 (TP-3) Nuclear Power Plant (NPP) with the Surry 1 & 2 NPPs for the reactor vessel (RV) closure head region of the control rod mechanism (CRM) nozzle penetrations; and the CRM nozzle inside diameter temper bead weld repair. The applicability will be accomplished by a comparison study that includes documenting the engineering data from both TP-3 and Surry NPPs, such as: applicable dimensions of features, materials, and plant operational transients to include time, temperature and pressure.

Both TP-3 and Surry 1 & 2 are Westinghouse Electric Co. pressurized light water reactors (PWR), 157 fuel assemblies, with "3-Loop" steam generator reactor coolant systems. The results of this comparative study of the critical parameters will show that the plants are nearly identical and that the engineering analyses performed for TP-3 are applicable to Surry. The results of this study are provided in the body of this report.

Introduction:

In order to demonstrate that the engineering analyses performed for the Turkey Point 3 NPP control rod drive mechanism nozzle inside diameter temper bead weld repair are applicable to Surry, a list of applicable parameters for each plant will be tabulated and compared. The list of parameters will include all features that are pertinent to the engineering analyses. Some typical parameters are the dimensions of the RV Closure Head radius, the number of CRM penetrations and spacing in the Closure Head, materials, and plant operational transients to include time, temperature and pressure.

Operating Transients Data:

The Framatome ANP Turkey Point 3 transients (Ref. 11, Appdx A) were compared with the transients submitted by Dominion Generation for Surry. The results of the transients bounding cases are given in Ref. 9. The results of the comparison concluded that the TP-3 transients bounding cases also bounded the transients listed in Table 1.

Engineering Analyses Parameters:

A number of pertinent engineering analysis data are contained in Tables 1, 2, and 3. These data are considered necessary to perform the various analyses. The components' dimensions/data provided or confirmed by Dominion Generation (Ref.s 1, 9, 10, 16, 22 through 31) were compared with the TP-3 data and are found to be acceptable.

Conclusion:

Based on the comparisons of Surry drawings and referenced engineering data received from Dominion Generation – Surry NPP, and TP-3 drawings and referenced engineering data, the engineering analyses for the CRM Nozzle ID Temper Bead Repair components for TP-3 are directly applicable to Surry 1 & 2 NPPs.

Record of Revision: Rev. 01 – See Page 5, Reference 18, removed reference to 32-5014129-01, reference to 32-5014129-00 is still applicable to this reconciliation document. Removed Ref. 19 as it is not used in Rev. 00 or 01. The Conclusions stated above and as in Rev. 00 of this document remains unchanged by this rev. Only Pages 1 and 5 are affected by Rev. 01. Oct. 31, 2001

* ALSO INCLUDES: Appdx A pg.s 1-10, Appdx B pg.s 1-2, Appdx C pg.s 1-25, Appdx D pg.s 1-15. Total Page Count = 58.

Table 1 RCS SPECIFICATIONS

Component	Turkey Point 3 Analyses (TP-3) Data Description	Reference Source	Surry Data Description	Reference Source
RCS Spec.s				
Design Conditions				
Design Pressure	2500 psia	Ref. 12, para. 3.15	2485 psig (2500 psia)	Ref. 1, Attmt 1-1, para. 1.1.2
Design Temperature	650 F	Ref. 12, para. 3.17	650 F	Ref. 1, Attmt 1-1, para. 1.1.2
Hydrotest Pressure	3125 psia	Ref. 12, Appdx B	3107 psig (3122 psia)	Ref. 1, Attmt 1-1, para. 1.1.2
Hydrotest Temperature			NDTT +60 F min.	Ref. 1, Attmt 1-1, para. 1.1.2
Hydrotest Temperature at Mfr			110 F	Ref. 1, Attmt 1-1, para. 1.1.2
Operating Conditions				
Coolant Fluid			Pressurizer Water	Ref. 1, Attmt 1-1, para. 1.1.3
Operating Pressure	2250 psia	Ref. 12, para. 3.16	2235 psig (2250 psia)	Ref. 1, Attmt 1-1, para. 1.1.3
Normal Operating Temperature	594 F	Ref. 12, Appdx B	543 F	Ref. 1, Attmt 1-1, para. 1.1.3
Inlet Temperature			543 F	Ref. 1, Attmt 1-1, para. 1.1.3
Outlet Temperature at Normal Temp.			605.8 F	Ref. 1, Attmt 1-1, para. 1.1.3
Initial Operating Limitations/Transients				
Heat Up and Cool Down Transients	200 HU and 200 CD Cycles, 5 Hydrotest Cycles at 2500psia at Operating Temp. and 1 cycle at 3125 psia at 100 F.	Ref. 11, Table 5.1, Ref. 12	The heating and cooling rate is limited to maximum 100 F per Hour. These rates will be safe for 200 Occurrences each. Thus, when starting at an isothermal condition at 100 F, the maximum heating rate is not to exceed 100 F per Hour up to operating temperature and, when starting at an isothermal condition at operating temperature, the maximum cooling rate is not to exceed 100 F per Hour returning to 100 F.	Ref. 9
			Plant Heatup at 100 F/Hr., 200 Occurrences, Normal Operating Condition; Plant Cooldown at 100 F/Hr., 200 Occurrences, Normal Operating Condition.	Ref. 9
Plant Loading and Unloading Transient	14,500 Cycles	Ref. 11, Table 5.1, Ref. 12	Plant Loading and Unloading at 5% Full Power per Minute, 29,000 Occurrences each at Normal Operating Condition. A total of 14,500 Cycles.	Ref. 9
Bounding of Remaining Transients Including:	2,800 Total Cycles	Ref. 11, Table 5.1, Ref. 12	2,800 Total Cycles	Ref. 9
10% Step Decrease	2,000 Cycles	Ref. 11, Table 5.1, Ref. 12	10% Step Load Increase and Decrease of Full power, 2,000 Occurrences,Normal Op. Cond.	Ref. 9
10% Step Increase		Ref. 11, Table 5.1, Ref. 12		
Large Step Decrease	200 Cyles	Ref. 11, Table 5.1, Ref. 12	Large Step Decrease, 200 Occurences, Normal Op. Cond.	Ref. 9
Loss-of-Load	80 Cycles	Ref. 11, Table 5.1, Ref. 12	Loss-of-Load, 80 Occurrences, Upset Condition	Ref. 9
Loss-of-Flow	80 Cycles	Ref. 11, Table 5.1, Ref. 12	Loss-of-Flow, 80 Occurrences, Upset Cond.	Ref. 9
Reactor Trip	400 Cycles	Ref. 11, Table 5.1, Ref. 12	Reactor Trip from Full power, 400 Occurences, Upset Cond.	Ref. 9
Loss-of-AC Power ,Trips, Step Changes, Etc.	40 Cycles	Ref. 11, Table 5.1, Ref. 12	Loss of Power, 40 Occurrences, Upset Cond.	Ref. 9

Table 2 REACTOR VESSEL CLOSURE HEAD ASSEMBLY

Component	Turkey Point 3 Analyses (TP-3) Data Description	Reference Source	Surry Data Description	Reference Source
CLOSURE HEAD ASSEMBLY				
Dry Weight			111,347 Lb.	Ref. 1, Attmt 1-4, para. 1.1.7
Closure Head Forging	184 in. OD x 2 Ft. 11-11/32 in. Length	Ref. 8, Part No. 51	15-Ft. 4 in. OD x 2-Ft. 11-11/32 in. Length	Ref. 30 & 31
Material				
Material	ASTM A-508, Class 2, Mn-Mo Steel, ASME Code Case 1332-2	Ref. 2, Part No. 51	ASTM A-508, Class 2, Mn-Mo Steel.	Ref. 22 & 23
Closure Head Plate	79-1/4 in. Inside Radius to basemetal x 6-3/16 in. min. thkns plus).156 min. Thkns cladding - SST.	Ref. 7, Part No. 50	79-1/4 in. Inside Radius to basemetal x 6-3/16 in. min. thkns plus).156 min. Thkns cladding - SST.	Ref. 28 & 29
Material (See Note 1 Below)	ASME SA-302, Grade B, Mn-Mo Steel	Ref. 2, Part No. 50	ASTM A-533, Grade B, Class 1, Mn-Mo Steel.	Ref. 22 & 23

Note 1 - An evaluation was performed to compare the material properties of SA-302 and SA-533. A review of Ref. 18, page 6, and Ref. 21, Page 9 demonstrates that the pertinent material properties at temperature are identical or nearly the same values that no significant difference would affect the results of the applicable stress analyses (Ref. 11 & 18).

Table 3 CONTROL ROD MECHANISM HOUSINGS

Component	Turkey Point 3 Analyses (TP-3) Data Description	Reference Source	Surry Data Description	Reference Source
Control Rod Mechanism Housing			Housing weldment consists of threaded 6-in. OD Adapter, and a 4-in. OD Body. Housing has an interference fit with the Closure Head and welded into the inside of the Closure Head with weld deposited Inconel.	Ref. 26 & 27
Quanity	65	Ref. 5, View: Key Plan	65	Ref.23 & 24
Spacing			8.466 in. centers	Ref.28 & 29
Material - CRM Adapter	ASME SA-182, Type 304, SST	Ref. 2, Part No. 1	ASME SA-182, Type 304, SST	Ref.23 & 24
Material - CRM Body	ASME SB-167 Inconel	Ref. 2, Part No. 2 - 14	ASME SB-167 Inconel	Ref.23 & 24
Vent Pipe			Nominal 1.00 in. Dia. Penetration.	Ref. 1, Attmt 3-4, para. 3.1.3
3-D FE Model Parameter List of CRM Housing (See Ref. 18 for Description of Parameters)				
thead	6+3/16 in.	Ref. 7	6.188 in.	Ref. 24 &25
tclad	0.156 in.	Ref. 7	0.156 in.	Ref.24 & 25
rbase	79+3/32+0.156 in.	Ref. 7	79+3/32+0.156 in.	Ref. 30 & 31
Rad To Noz (Max.)	53.544 in.	Ref. 5	53.544 in.	Ref. 30 & 31, Top View, calc'd value.
DiaPen	4.000 in.	Ref. 5	4.000 in.	Ref. 30 & 31, Detail for Hole No. 1, and Detail for All Adapter Holes Except Hole No. 1.
tButter	0.25 in	Ref. 5	0.25 in.	Ref. 30 & 31, Detail for Hole No. 1, and Detail for All Adapter Holes Except Hole No. 1.
WPirad	.5-tButter	Ref. 5	.5-tButter	Ref. 30 & 31, Detail for Hole No. 1, and Detail for All Adapter Holes Except Hole No. 1.
WldAngl	20 degrees	Ref. 5	20 degrees	Ref. 30 & 31, Detail for Hole No. 1, and Detail for All Adapter Holes Except Hole No. 1.
NozOD	4.025 in.	Ref. 5	4.025 in.	Ref. 26 & 27
NozTw	0.6375 in.	Ref. 5	0.6375 in.	Ref. 26 & 27

REFERENCES

Reference No.	Document No.	Description	Source
1*	78-S25 .	Final Design Surry Power Station, Part Length Control Rod Removal,Rev. 2, dated 7/18/80, Attachment.	Dominion Generation, Surry Power Station, Facsimile Transmittal, dated 10/5/2001, To: Alvin McKim - FRA-ANP, From: Doug Lawrence - Dominion -Surry, 25 Pages, time 13:05 hrs.
2	02-117877E, Rev. 5	Material List, Reactor Vessel, Westinghouse Atomic Power Div., Contr No. 610-0116-51 & 52	FRA-ANP Records Center, Lynchburg, VA
3	02-117878E, Rev. 5	Closure Head Assembly, Contr. No. 610-0116-52	FRA-ANP Records Center, Lynchburg, VA
4	02-117880E, Rev. 5	Detail & Sub-Assy, Control Rod Mech. Housing, Contr. No. 610-0116-52	FRA-ANP Records Center, Lynchburg, VA
5	02-117881E, Rev. 6	Closure Head Sub-Assembly, Contr. No. 610-0116-52	FRA-ANP Records Center, Lynchburg, VA
6	02-5012151E, Rev. 5	CRDM Nozzle ID Temperbead Weld Repair Boring Option B&W 177 FA Plants, dated 8/3/01.	FRA-ANP Records Center, Lynchburg, VA
7	02-88181C, Rev. 1	Closure Head Center Disc, Contr. No. 610-0116-52	FRA-ANP Records Center, Lynchburg, VA
8	02-117883E, Rev. 1	Details Closure Head Flange, Contr. No. 610-0116-52	FRA-ANP Records Center, Lynchburg, VA
9*	N/A	Surry Reactor Head Inspection - Design Information Transmital	Dominion Generation, Letter From Dean I. Price To: Paul Ulmer of FRA-ANP, dated Oct. 12, 2001.
10*	676500 Rev. 1	Equipment Specification, dated 4/29/71, "Addendum to Equipment Spec. 676413, Rev. 1, Project: SurryPower Station II, Eqpt: Reactor Vessel, System: Reactor Coolant.	Dominion Generation, Facsimile Transmittal, dated 10/12/2001, To: Paul Ulmer/Jim Dorman- FRA-ANP, From: Dean Price, 10 Pages, time 09:54 hrs.
11	32-5014640-00	Turkey Point - CRDM Temperbead Bore Weld Analysis	FRA-ANP Records Center, Lynchburg, VA
12	51-5014575-00	Turkey Point CRDM Noz. ID Temper Bead Weld Repair Reqmts	FRA-ANP Records Center, Lynchburg, VA
13	Not Used		
14	Not Used		
15	Not Used		
16*		Surry Reactor Head Inspection - Design Information Transmital	Dominion Generation, Letter From: Dean Price, To: Paul Ulmer- FRA-ANP, Subject - Surry Reactor Head Inspection, Design Information Transmittal, dated 10/17/2001.
17	32-5015219-00	Surry CRDM Noz IDTB Weld Anomaly Flaw Eval.	FRA-ANP Records Center, Lynchburg, VA
18	32-5014129-00	TP CRDM Conn. 3D FE Model	FRA-ANP Records Center, Lynchburg, VA
19	Not Used		
20	32-5015220-00	Surry CRDM Noz IDTB J-Groove Weld Flaw Eval.	FRA-ANP Records Center, Lynchburg, VA
21	32-5011864-00	CRDMH Connection 3D FE Model	FRA-ANP Records Center, Lynchburg, VA
22	02-131174E, Rev. 3	Material List, Contr No. 610-0137-51 & 52	FRA-ANP Records Center, Lynchburg, VA
23	02-134804E, Rev. 5	Material List, Contr No. 610-0147-51 & 52	FRA-ANP Records Center, Lynchburg, VA
24	02-131180E, Rev. 1	Closure Head Details, Contr No. 610-0137-52	FRA-ANP Records Center, Lynchburg, VA
25	02-134810E, Rev. 1	Closure Head Details, Contr No. 610-0147-52	FRA-ANP Records Center, Lynchburg, VA
26	02-131177E, Rev. 3	Control Rod Mech. Housing, Contr No. 610-0137-52	FRA-ANP Records Center, Lynchburg, VA
27	02-134807E, Rev. 1	Control Rod Mech. Housing, Contr No. 610-0147-52	FRA-ANP Records Center, Lynchburg, VA
28	02-131175E, Rev. 1	Closure Head Assembly, Contr No. 610-0137-52	FRA-ANP Records Center, Lynchburg, VA
29	02-134805E, Rev. 0	Closure Head Assembly, Contr No. 610-0147-52	FRA-ANP Records Center, Lynchburg, VA
30	02-131178E, Rev. 3	Closure Head Sub-Assembly, Contr No. 610-0137-52	FRA-ANP Records Center, Lynchburg, VA
31	02-134808E, Rev. 1	Closure Head Sub-Assembly, Contr No. 610-0147-52	FRA-ANP Records Center, Lynchburg, VA

* These references are not in the Framatome ANP Records Center. The use of these Customer Supplied Documents for Surry CRDM Weld Repair, Contr. No. 4160048, and the design input data contained therein are approved by the Project Manager. PM Signature: P. M. Ulmer ____________.

APPENDICES: Customer Supplied Documents

Appendix A - Dominion Generation Letter, Subject: Surry Reactor Head Inspection Design Information Transmital, From Dean I. Price, To: Paul Ulmer of FRA-ANP, Dated Oct. 12, 2001.

Appendix B - Dominion Generation Letter, Subject: Surry Reactor Head Inspection Design Information Transmital, From Dean I. Price, To: Paul Ulmer of FRA-ANP, Dated Oct. 17, 2001.

Appendix C - Dominion Generation, Surry Power Station, Facsimile Transmittal, dated 10/5/2001, To: Alvin McKim - FRA-ANP, From: Doug Lawrence - Dominion -Surry, 25 Pages, time 13:05 hrs.

Appendix D – Westinghouse Electric Co., Facsimile Transmittal, dated 10/12/2001, To: Dean Price of Dominion Gen. Surry NPP, From Justin Ledger, 15 Pages.

Dominion Generation
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion

Framatome ANP, Inc
3315 Old Forest Road
Lynchburg, BA 24506-0935

Attention: Mr. Paul Ulmer

October 12, 2001

Subject: Surry Reactor Head Inspection
Design Information Transmittal

Dear Mr. Ulmer

Please find attached a Memorandum from our Engineering Mechanics department to myself concerning design information such as transients, operating cycles, etc that you have requested to be used in the engineering for a potential reactor head penetration repair should one be needed. If additional information is needed in this area, please contact me at 804-273-3586.

Dean I. Price

Dean I. Price
Project Engineer

bcc: A. McKim
B. De Cooman
R. Dorman
M. Carpenter
D. Matthews
M. Sloman
R. Smith

APPENDIX A
51-5015197-00

Dominion Memorandum

October 11, 2001

To:	D. I. Price
Company:	Dominion Resources Services, Inc.
Department:	Nuclear Projects Department, Civil/Mechanical
Location:	ITC-3NW
From:	D. R. McGowan
Company:	Dominion Resources Services, Inc.
Department:	Nuclear Engineering Department, Engineering Mechanics Group
Location:	ITC-3NW

Review of Framatome Transient Set for Surry CRDM Penetrations Analysis

Per your request, Engineering Mechanics (EM) has reviewed the transient data supplied by Framatome for the design of the Control Rod Drive Mechanisms (CRDMs) for Surry Units 1 and 2. The following comments apply.

The Surry reactor vessels (including the CRDM penetrations) are designed for the following thermal and pressure transient conditions (References 1 and 2):

1. Plant heatup at 100°F per hour, 200 occurrences, normal operating condition
2. Plant Cooldown at 100°F per hour, 200 occurrences, normal operating condition
3. Plant Loading at 5% of full power per minute, 29,000 occurrences, normal operating condition
4. Plant Unloading at 5% of full power per minute, 29,000 occurrences, normal operating condition
5. Step load increase of 10% of full power, 2000 occurrences, normal operating condition
6. Step load decrease of 10% of full power, 2000 occurrences, normal operating condition
7. Large step decrease in load (with steam dump), 200 occurrences, normal operating condition
8. Loss of load (without immediate turbine or reactor trip), 80 occurrences, upset condition
9. Loss of power (blackout with natural circulation in RCS), 40 occurrences, upset condition
10. Loss of flow (partial loss of flow – one pump only), 80 occurrences, upset condition
11. Reactor trip from full power, 400 occurrences, upset condition
12. Steam pipe break, 1 occurrence, faulted condition
13. Turbine roll test, 10 occurrences, normal operating condition
14. Primary side hydrostatic test before startup at 3105 psig, 5 occurrences, normal operating condition

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15. Primary side hydrostatic test at 2485 psig, 50 occurrences, normal operating condition

16. Steady state fluctuations, ∞ occurrences

Details of the review of Framatome's transients are discussed below. The number of occurrences for the transients assumed by Framatome are included in the Figures.

- For heatup, Framatome's heatup curve (Figure 1) shows a rate of 100°F/hr and a range of 100°F to 600°F. This heatup rate matches the design rate for Surry. The range bounds Surry's design range. For design purposes, an ambient temperature of 70°F was assumed, and the no-load RCS temperature is 547°F. Per Reference 4, the full power upper head mean fluid temperature for Surry is 597.8°F. Therefore, the heatup rate and range proposed by Framatome are judged to be bounding. Framatome's heatup pressurization curve (Figure 2) shows an approximate rate of 645 psig/hr. This number does not bound the design value of 740 psig/hr; however, it bounds the actual pressurization rates used during plant heatup.

- For cooldown, Framatome's cooldown curve (Figure 3) shows a rate of -100°F/hr and a range of 600°F to 100°F. This cooldown rate matches the design rate for Surry. The range bounds Surry's design range as discussed above. Framatome's cooldown pressurization curve (Figure 4) shows an approximate rate of -645 psig/hr. This number does not bound the design value of 740 psig/hr; however, it bounds the actual rates used during plant cooldown.

- For plant loading, the design basis for Surry is for 29,000 cycles, based on the assumption that the plant is operating in a load-follow mode. The Surry units do not operate in a load follow mode; thus, the number of cycles for this transient is very conservative. Per Reference 4, the temperature range for this transient would be 547°F to 597.8°F, and the transient would occur over a time period of 20 minutes (5% of full power per minute). The temperature range listed in Framatome's plant loading transient is 547°F to 618°F over 20 minutes (Figure 5). In all cases, the RCS pressure remains constant at 2235 psig (Figure 6). Framatome has assumed 14,500 cycles for this transient. The Framatome transient is bounding.

- For plant unloading, the design basis for Surry is for 29,000 cycles, again based on the assumption that the plant is operating in a load-follow mode. As discussed previously, the number of cycles for this transient is very conservative. Per Reference 4, the temperature range for this transient would be 597.8°F to 547°F, and the transient would occur over a time period of 20 minutes (5% of full power per minute). The temperature range listed in Framatome's plant loading transient is 618°F to 547°F over 20 minutes (Figure 7). In all cases, the RCS pressure remains constant at 2235 psig (Figure 8). Framatome has assumed 14,500 cycles for this transient. The Framatome transient is bounding.

- For the remaining transients of increasing temperatures, Framatome proposes 2800 occurrences of a transient from 577°F to 617°F (+40°F) in 10 seconds (Figure 9), accompanied by a rise in pressure from 2235 to 2585 psig (+350 psi) (Figure 10). For the remaining transients of decreasing temperatures, Framatome proposes 2800 occurrences of a transient from 617°F to 517°F (-100°F) in 10 seconds (Figure 11), accompanied by a drop in pressure from 2235 to 1735 psig (-500 psi) (Figure 12). Review of the 10% step increase, 10% step decrease, large step decrease in load (with steam dumps), loss of load, loss of flow, reactor trip, turbine roll, and loss of power design basis transients show that they are collectively bounded by the transients assumed by Framatome, both in magnitude and number of occurrences.

- For the hydrostatic pressure tests, one planned test to 3107 psi occurred during pre-operational testing. No additional testing is planned. Also, no additional testing above normal operating pressure is to be performed, as allowed by ASME Code Case N-498-1. Thus, the hydrotstatic test transients do not need to be considered.

References:

1. Equipment Specification 676499, Revision 1, dated 4/28/71, "Addendum to Equipment Specification 676413, Rev. 1, Project: Surry Power Station I, Equipment: Reactor Vessel, System: Reactor Coolant."

2. Equipment Specification 676500, Revision 1, dated 4/29/71, "Addendum to Equipment Specification 676413, Rev. 1, Project: Surry Power Station II, Equipment: Reactor Vessel, System: Reactor Coolant."

3. Calculation 30660-1130, "Reactor Vessel – Final Stress Report," Revision 1 (North Anna Units 1 and 2).

4. Engineering Transmittal NAF 95-162, Rev. 0, "Reactor Vessel Coolant Temperature Design Input for Use in Upper Head Penetration Inspection Program, Surry Power Station Units 1 and 2."

Prepared by: D. M[illegible] Date: 10-11-01

Reviewed by: K. K. Dwivedy Date: 10-11-01

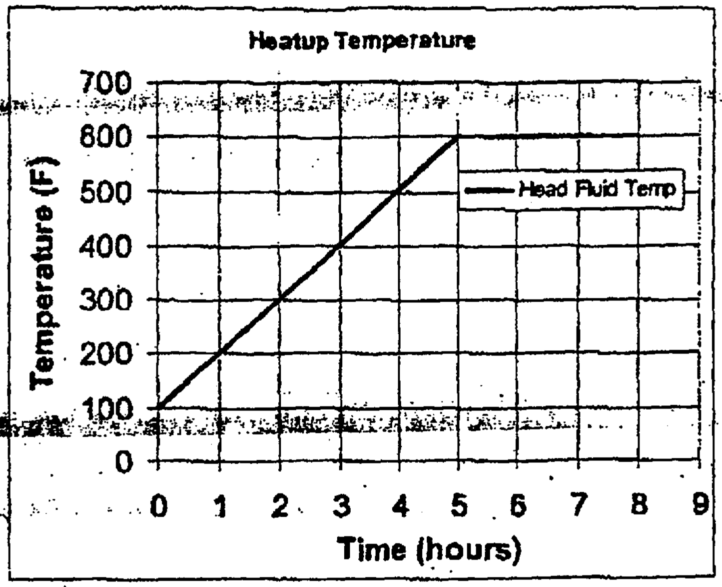


Figure 1

Occurrences = 200

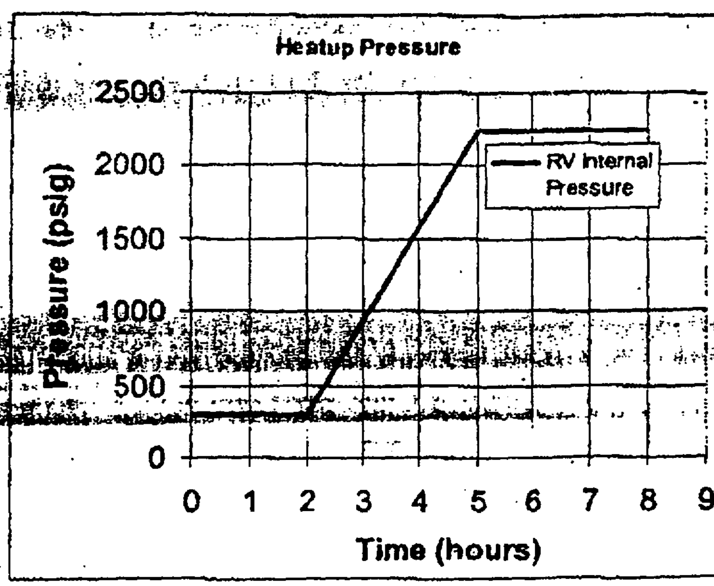


Figure 2.

Occurrences = 200

51-5015197-00

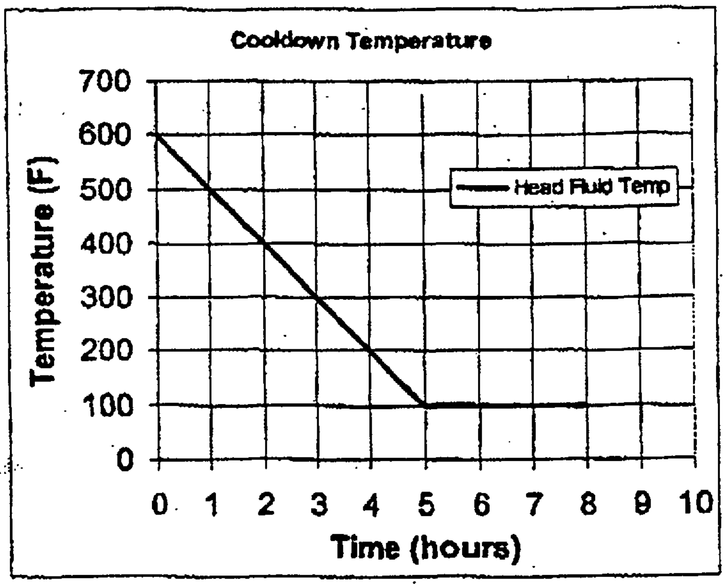


Figure 3

Occurrences = 200

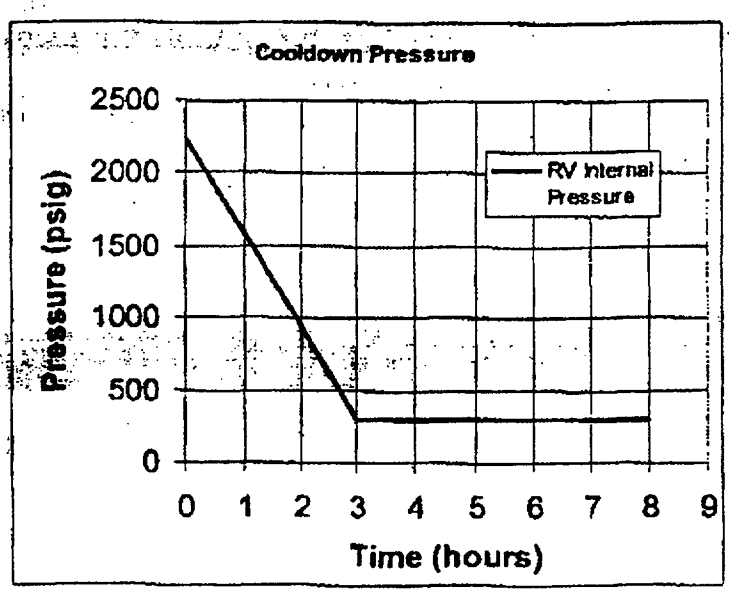


Figure 4.

Occurrences = 200

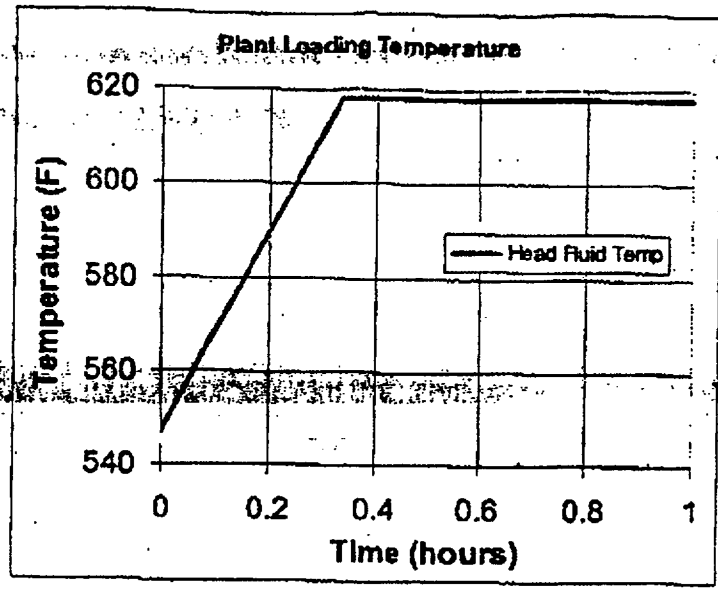


Figure 5

Occurrences
= 14,500

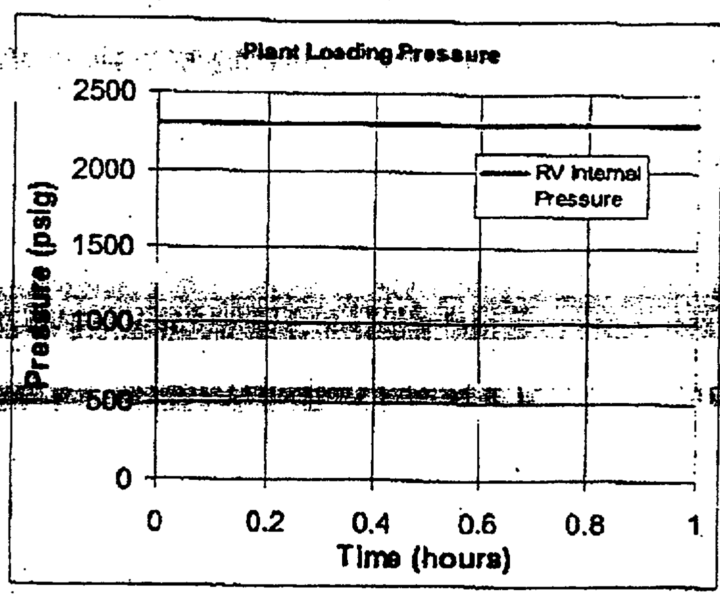


Figure 6

Occurrences
= 14,500

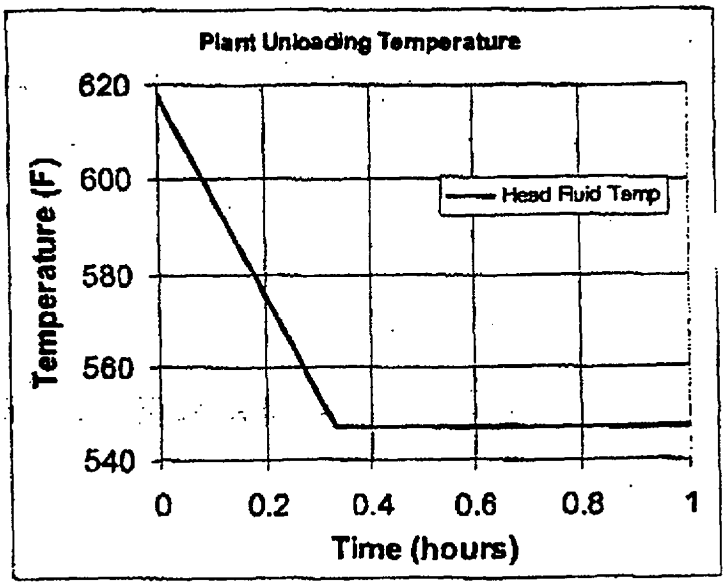


Figure 7

Occurrences = 14,500

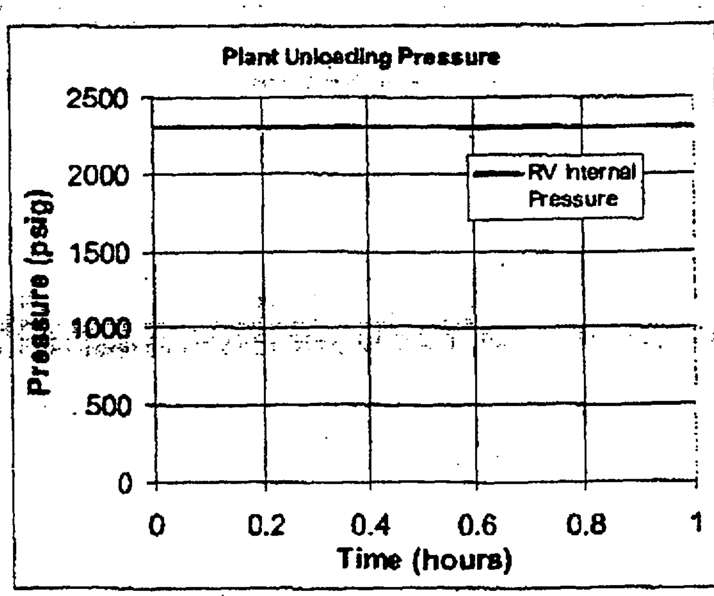


Figure 8

Occurrences = 14,500

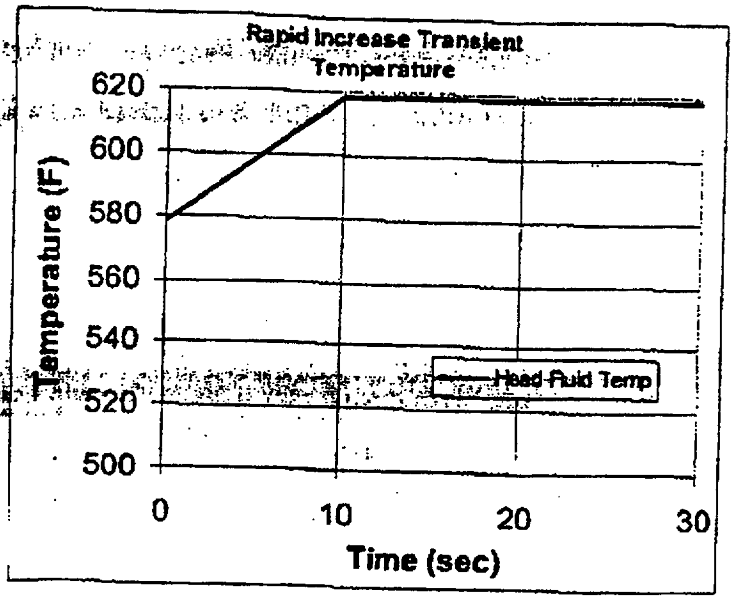


Figure 9

Occurrences
= 2800

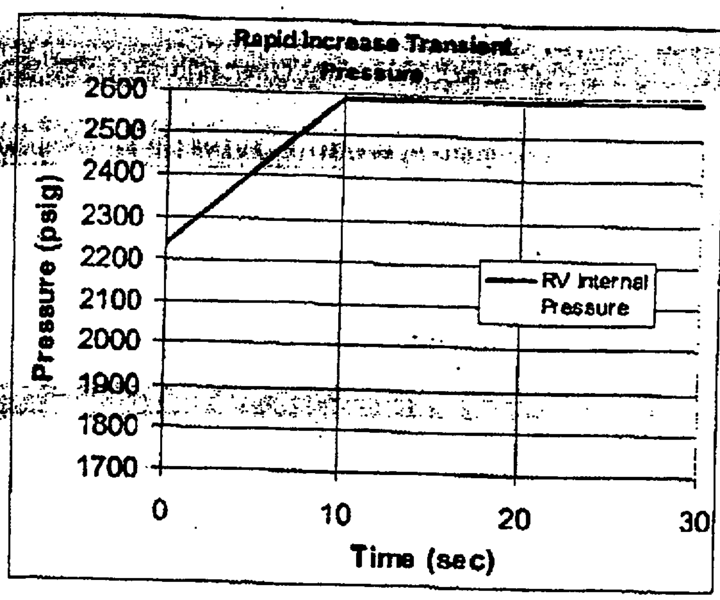


Figure 10

Occurrences
= 2800

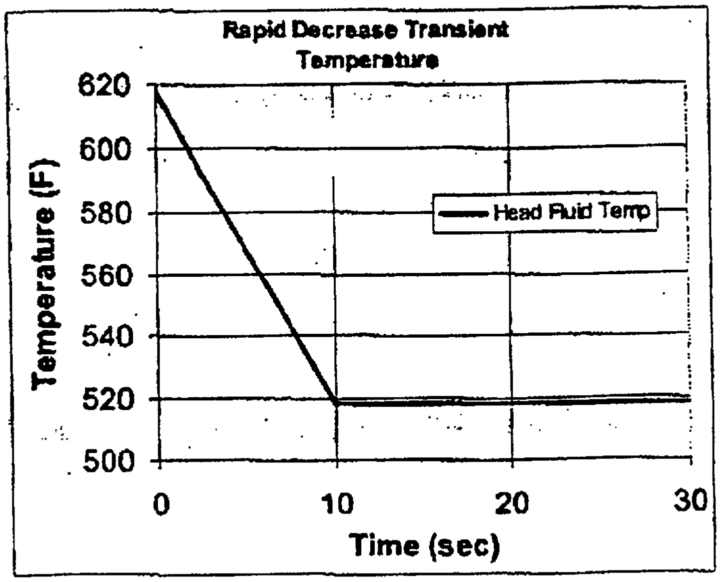


Figure 11

Occurrences
= 2800

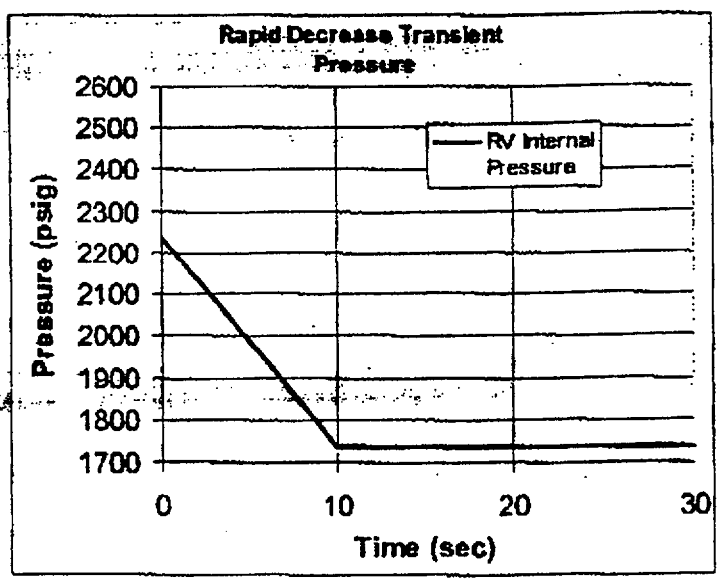


Figure 12

Occurrences
= 2800

51-5015197-00

Dominion Generation
Nuclear Engineering
Innsbrook Technical Center
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion

APPENDIX B
Doc. Id 51-5015197-00
Page 1 of 2

Framatome ANP, Inc
3315 Old Forest Road
Lynchburg, BA 24506-0935

October 17, 2001

Attention: Mr. Paul Ulmer

Subject: Surry Reactor Head Inspection
Design Information Transmittal

Dear Mr. Ulmer:

Attached to this letter are the highlighted drawings that Framatome sent to Dominion for design information verification with the corresponding Westinghouse information. This information has been verified with the exceptions listed below (which were sent to Framatome in an earlier e-mail) and so indicated with additional highlighting next to the requested information. This information can be used as design input for the Surry Units 1 and 2 Reactor Vessel Head Repair.

Exceptions:

1. Drawing 131175E--I can't verify the original material thickness of 6 9/16" for the head.

2. Drawing 131174E--I have not been able to verify notes 2, 3, 4, 5, 6, 9, 11, 12. I'm still working on this. Also I have not confirmed the appreciable stress due to bolting. Our engineering mechanics guys think this is a good assumption but we will have the stress report on Thursday and will verify this.

3. Drawing 131178E--Cannot verify Westinghouse weld procedures are the same as Framatome's. The NDE requirements are the same as far as calling for a PT.

4. Drawing 131177E--Section "Machining of Control Rod Mechanism Housing" shows 2 blocks at the right end of the housing. I can verify the left block and everything in the right block except the last word or number. It is also unclear on the drawings that Westinghouse has. They said that it is "/308" but that really doesn't seem to make any sense.

5. Drawing 134809E--Section 15--I'm not sure what is meant by "2" dia (and then a triangle)" but I have not been able to verify this.

6. Drawing 131179E--There are a couple of areas circled on this drawing and they appear to be head vent piping details. I have verified that the Unit 2 drawings agree with the Westinghouse drawings but I can't read your unit 1 details. I am assuming that these are the same as the unit 2 details.

7. Drawing 5015107D--Most of these dimensions have been verified and a couple are fractionally different and are listed on the marked up drawing.

8. Additional information was requested on CRDM housing material and welding. This is listed below with the response in bolded type.

As part of your design input response letter can you please confirm that the following materials are applicable to the Surry 1 and 2 CRM penetrations?

1) CRM Housing Nozzle = SB-167 (Inconel). **Correct**

2) Closure Head Cladding = Austenitic Stainless Steel, Type 316. **It is austenitic stainless but I have not been able to verify the 316. All of the Westinghouse specs say "304 or better".**

3) Closure Head/CRM Housing Nozzle, J-Groove weld buttering = Alloy 600 (Inconel). **According to our welding experts, the weld material comparable to Inconel 600 is Inconel 82/182. According to them, Inconel 600 is not a weld filler material.**

4) Closure Head/CRM Housing Nozzle, J-Groove weld filler metal = Alloy 600 (Inconel). **See item 3 response.**

If you have any additional questions or need any more information please do not hesitate to call me at 804-273-3586

Dean I. Price
Project Engineer

APPENDIX B
Doc. Id. 51-5015197-00
Page 2 of 2

APPENDIX C
Doc. Id. 51-5015197-00

Ref. 1

Dominion
Generation
Surry Power Station

FACSIMILE TRANSMITTAL

TO: AL McKIM / PAUL ULMER
PHONE:
FAX:

FROM: DOUG LAWRENCE
PHONE: (757) 365-2755
FAX: - 2750

E-MAIL :

DATE: 10/5/01 TIME: 1305

OF PAGES 25 (INCLUDING THIS PAGE)

MESSAGE:

HERE IS PART LENGTH CONTROL ROD AND VESSEL MATERIAL & DESIGN DATA. WILL FAX STRESS REPORT NEXT.

Doug

51-5015197-00

Ref I

898.15 A

FINAL DESIGN
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

TO: SUPERVISOR – ENGINEERING SERVICES 1	DESIGN CHANGE NO: 78-S25 2
TITLE: Part Length Control Rod Removal 3	UNIT NO: 1 & 2 4

FINAL DESIGN: 5

(FINAL DESIGN SHALL CONSIST OF: 1. REFERENCES; 2. DESCRIPTION; 3. DRAWINGS; 4. DESIGN BASIS; 5. OPERATIONAL REQUIREMENTS; 6. PERIODIC TEST REQUIREMENTS; 7. MATERIALS LIST AND 8. EQUIPMENT SPECIFICATIONS.)

FINAL DESIGN DEVELOPED BY: Lawrence Lobo *Lawrence Lobo* 6	COMPLETION DATE: 3/20/79 7
PROJECT ENGINEER: Lawrence Lobo *Lawrence Lobo* 8	DATE: 3/20/79 9
REVIEWED BY DESIGN CONTROL ENGINEER: R. H. Coupe [signature] 10	DATE: 4-4-79 11
REVIEWED BY SUPERVISOR-ENGINEERING SERVICES: D. A. Christian *David A Christian* 12	DATE: 4-9-79 13
REVIEWED BY SUPERVISOR-NUCLEAR ENGR. SERVICES: [signature] 14	DATE: 10-1-79 15
REVIEWED BY STATION NUCLEAR SAFETY AND OPERATING COMMITTEE: CHAIRMAN'S SIGNATURE: *J. Wilson* 16	DATE: APR 9 1980 17

PROJECT AUTHORIZATION (ATTACH, IF REQUIRED.) 18

[X] REQUIRED; NO.: Surry No. 1 81216406 / Surry No. 2 81216506 [ ] NOT REQUIRED

REVISIONS TO FINAL DESIGN (ATTACH "FIELD CHANGE"): 19

REVISION NUMBER:	1	2										
DATE:	4/25/80	7/10/80										

REMARKS: 20

51-5015197-00 Pg 2

688.10

FINAL DESIGN (SUPPLEMEN
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

ATTACH TO: FINAL DESIGN 1	DESIGN CHANGE NO. 2
	78-S25

FINAL DESIGN (CONTINUED): 3

1.0 REFERENCES:

1.1 Royal Industries, Model 121 J001 Part Lenth Control Rod Drive Manual.

1.2 MMP-C-RC-035

1.3 OP-4.5

1.4 Vepco Quality Assurance Manual, Section 3

1.5 FSAR Section 3

1.6 W FS-78-1, Rev. October 18, 1978

2.0 DESCRIPTION:

2.1 Description of the anti-rotation devices can be found in the Westinghouse proposal for the Removal of Part Length Control Rods dated April 25, 1978. A copy is attached for reference.

3.0 DRAWINGS:

3.1 The appropriate drawings are attached.

3.2 Figure 1: Partial Length Anti Rotation Housing

Figure 2: Partial Length Up Position Leadscrew Clamp

Figure 3: Partial Length Conoseal Assembly

Figure 4: Partial Length Up Position Lead Screw Retainer

Figure 5: Locations of P/L Control Rods

4.0 DESIGN BASIS:

4.1 The intent of the Part Length Control Rods was to control axial power distribution and to suppress xenon oxcillations.

4.2 The utilization of Part Length Control Rdds for axial power distribution is not desirable. The insertion of the Part Length Control Rods would cause the lowering of power in the axial region just below and above the neutron absorbing material of the Part Length Control Rod.

4.3 At the time the Surry Units were designed, there was no stringent restriction on $\Delta\phi$ band. At the present time, there is a restriction on maintaining a narrow $\Delta\phi$ band of ± 5% which reduces xenon oscillations to a very low level.

888.16

FINAL DESIGN (SUPPLEME)
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

ATTACH TO: FINAL DESIGN 1	DESIGN CHANGE NO. 2
	78-S25

FINAL DESIGN (CONTINUED): 3

4.0 DESIGN BASIS: (CONTINUED)

4.4 Technical Specifications for Surry Power Station Units 1 and 2 do not allow the use of the part length control rods during operation. Westinghouse's study on part length control rod removal and operational experience in Surry indicate that the removal of the part length control rods is desirable.

5.0 OPERATIONAL REQUIREMENTS:

5.1 The reactor coolant system is to be at refueling shutdown condition in accordance with the plant technical specifications.

5.2 Once the part length control rods are removed, additional operational requirements are not necessary.

6.0 PERIODIC TEST REQUIREMENTS:

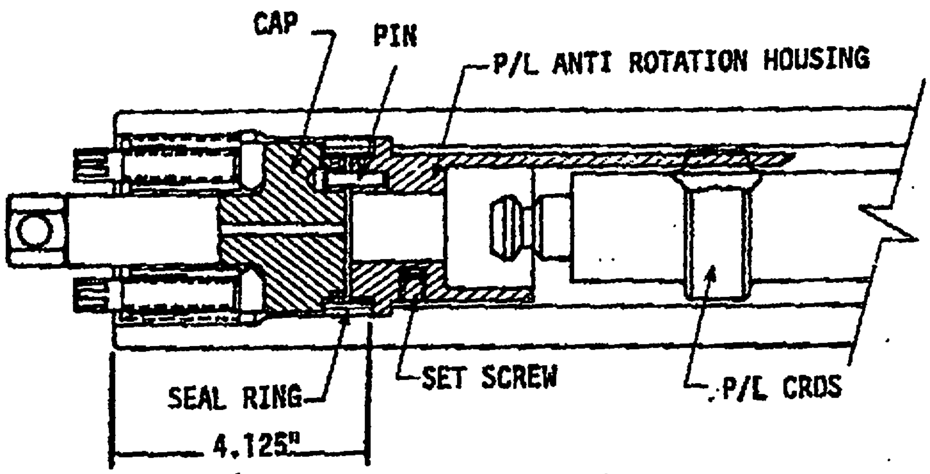
6.1 After the part length control rods are removed, the seals at the top of the part length lead screw travel housing need never be opened during a refueling. Since the seal is never broken, any possibility of leakage during plant startup following an outage is virtually eliminated. Therefore, there is no need for periodic testing.

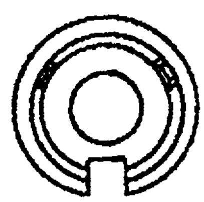
7.0 MATERIALS LIST:

7.1 See Westinghouse proposal dated April 25, 1978 attached.

8.0 EQUIPMENT SPECIFICATIONS:

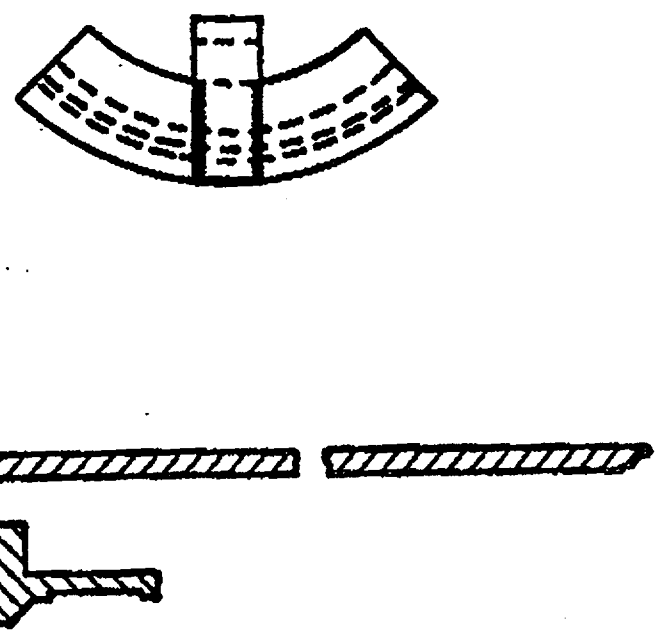
8.1 Not required





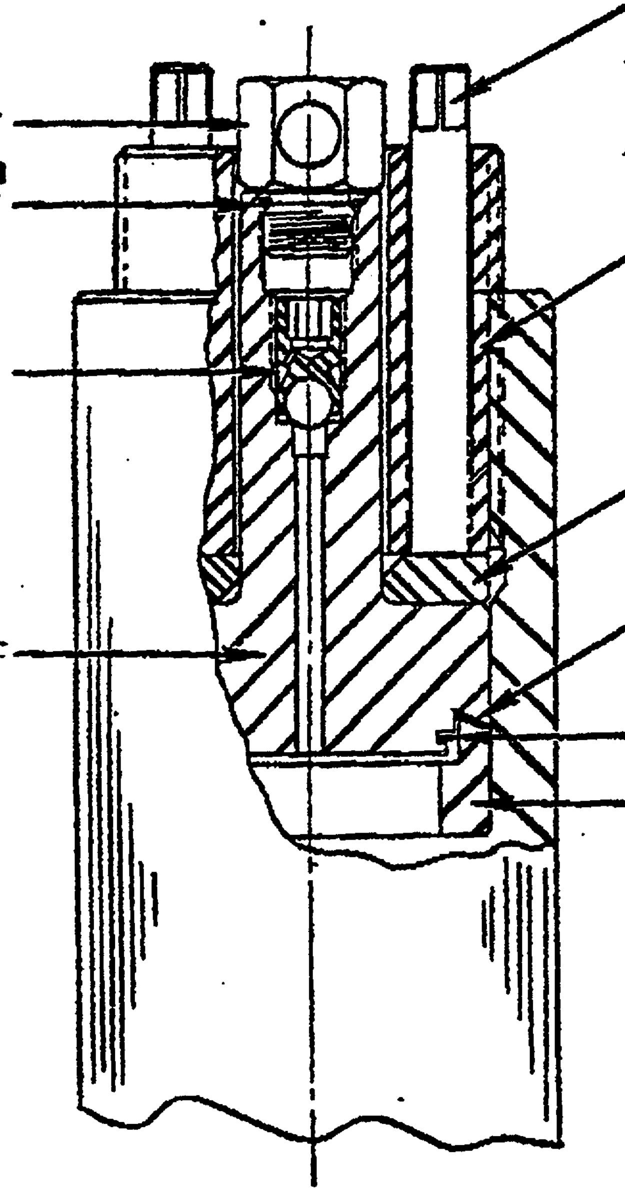
PARTIAL LENGTH ANTI ROTATION HOUSING

FIGURE 1



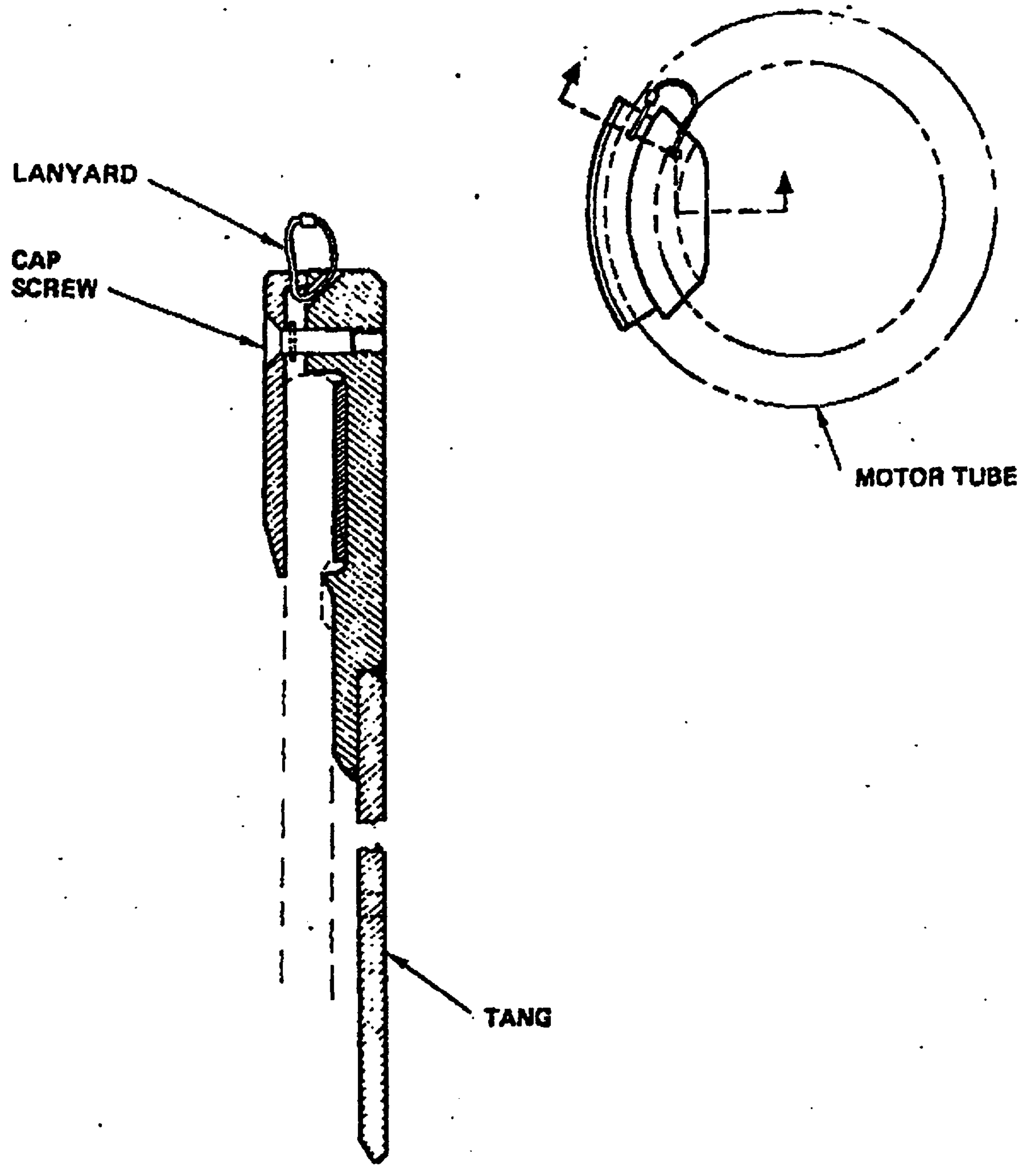
PARTIAL LENGTH UP POSITION LEADSCREW CLAMP

FIGURE 2



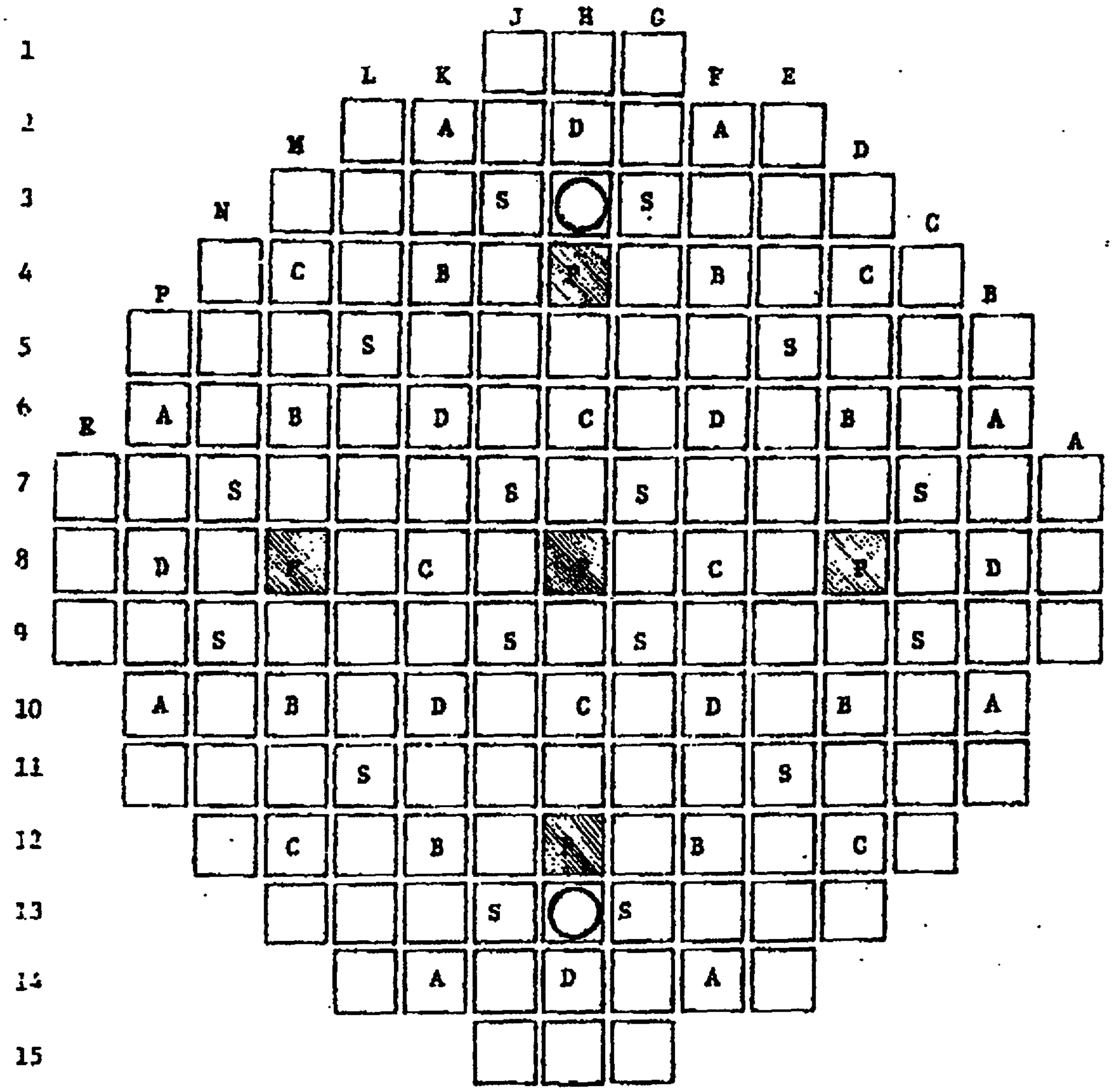
PARTIAL LENGTH CONOSEAL ASSEMBLY

FIGURE 3



PARTIAL LENGTH UP POSITION LEADSCREW RETAINER

FIGURE 4



CONTROL ROD ASSEMBLY BANKS

Function	Number of Assemblies
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown (S)	16
Part Length (P)	5
	53

◯ = SOURCE ASSEMBLY LOCATIONS

FIGURE 5: LOCATIONS OF PART LENGTH CONTROL RODS

No. 888.19A

FINAL DESIGN IMPLEMENTATIO AND TESTING
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

TO: 1	DESIGN CHANGE NO: 2 78-S25
DESIGN CHANGE TITLE: 3 PART LENGTH CONTROL ROD REMOVAL	UNIT NO: 4 1
DESIRED IMPLEMENTATION DATE: 5 DURING STEAM GENERATOR REPLACEMENT	
FINAL DESIGN CONTROLLING PROCEDURE: 6 PROCEDURE SHALL CONSIST OF: 1. PURPOSE; 2. INITIAL CONDITIONS; 3. PRECAUTIONS; 4. INSTRUCTIONS; ☒ COPY ATTACHED	
FINAL DESIGN TESTING: 7 PROCEDURE SHALL CONSIST OF: 1. PURPOSE; 2. INITIAL CONDITIONS; 3. PRECAUTIONS; 4. INSTRUCTIONS; 5. ACCEPTANCE CRITERIA. COPY ATTACHED: ☐ MECHANICAL TESTING ☐ ELECTRICAL TESTING ☐ INSTRUMENT TESTING ☐ CHEMICAL TESTING	
FINAL DESIGN CONTROLLING AND TESTING PROCEDURES: 8	
SUBMITTED BY PROJECT ENGINEER: 9 Lawrence Lobo *Lawrence Lobo*	DATE: 10 7-18-80
REVIEWED BY DESIGN CONTROL ENGINEER: 11 Richard Coupe *[signature]*	DATE: 12 7-18-80
RECOMMENDED APPROVED BY SUPERVISOR-ENGINEERING SERVICES: 13 David Fortin *[signature]*	DATE: 14 7-21-80
REVIEWED BY QUALITY CONTROL: 15 Frank Rentz *[signature]*	DATE: 16 7-21-80
APPROVED BY STATION NUCLEAR SAFETY AND OPERATING COMMITTEE: 17 CHAIRMAN'S SIGNATURE: *JLWilson*	DATE: 18 8/1/80

REMARKS: 19

This procedure addendum inserted as Field Change #2

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888.9 A (SURRY)

DESIGN CHANGE REQUEST
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

TO: SUPERVISOR - ENGINEERING SERVICES 1	DESIGN CHANGE NO.: 78-525 2
SYSTEM: Reactor Control 3	COMPONENT TAG NO.: Part Length Control Rods 4
UNIT NO. 1 & 2 5	

REFERENCES: Letter (??) to VEPCO (SYLVIA) 4/25/78 - "Removal of Part Length Control Rods - Surry Units One & Two" 6

BRIEF DESCRIPTION OF CHANGE REQUESTED (ATTACH ADDITIONAL PAGES, IF REQUIRED): 7
Remove P/L RCC from core - replace with "thimble plug" inserts. Install anti-rotation device to hold lead screw up in head.

REASON FOR CHANGE: 8
1) Decreased outage time during refueling outage - 2) Decreased radiation exposure - 3) Operation not allowed by T/S.

CHANGE REQUESTED BY: G. Kane 9 | DATE: 7/13/78 10

REVIEWED BY: 11
COGNIZANT SUPERVISOR: J W Wilson | DATE: 7-20-78 12

RECOMMENDED ACTION: 13
☒ APPROVED ☐ DISAPPROVED ☐ APPROVED AS MODIFIED

PROJECT ENGINEER: L. LOBO 14 | DATE ASSIGNED: 11/1/78 15 | DATE REQUIRED: 16

☒ ENGINEERING REVIEW ATTACHED 17

QUALITY GROUP CLASSIFICATION: ☒ A ☐ B ☐ C ☐ IE ☐ I ☐ NC ☐ D ☐ E ☐ OTHER 18

19 TECH SPEC. ITEMS: ☐ NO ☒ YES SECT. NO. 3.12 20

IMPLEMENTATION METHOD: ☒ DESIGN CHANGE PROGRAM ☐ MAINTENANCE PROGRAM MAINTENANCE REPORT NO. 21

PROJECT ENGINEER'S SIGNATURE: Lawrence Lobo 22 | DATE: 11/1/78 23

☒ SAFETY ANALYSIS ATTACHED (REQ'D FOR SAFETY-RELATED ITEMS) 24

25 TECH SPEC. CHANGE REQUIRED ☒ YES ☐ NO 26

UNREVIEWED SAFETY QUESTION ☐ YES ☒ NO 27

PROJECT ENGINEER'S SIGNATURE: Lawrence Lobo 28 | DATE: 11/1/78 29

DESIGN CONTROL ENGINEER'S RECOMMENDED ACTION: ☒ APPROVE ☐ DISAPPROVED 30

APPROVAL LEVEL ☒ NRC LEVEL ☐ SYSTEM LEVEL ☐ STATION LEVEL 31

METHOD OF IMPLEMENTATION ☒ DESIGN CHANGE PROGRAM ☐ MAINTENANCE PROGRAM

DESIGN CONTROL ENGINEER'S SIGNATURE: DHKaufle 32 | DATE: 11/29/78 33

SUPERVISOR - ENGINEERING SERVICES' REVIEW: Tech. Spec. revision to be handled as a portion of the core reload package 34

☒ APPROVED ☐ DISAPPROVED ☒ APPROVAL LEVEL VERIFIED

☒ STATION TO COMPLETE FINAL DESIGN ☐ PRODUCTION SERVICES RESPONSIBLE FOR FINAL DESIGN

PROJECT AUTHORIZATION ATTACHED (IF REQUIRED): 35

NOT REQ'D. ☒ REQ'D PRIOR TO FINAL DESIGN IR 99-0457 REQ'D POST FINAL DESIGN

SUPERVISOR ENGINEERING SERVICES' SIGNATURE: T W Mueller 36 | DATE: 11/30/78 37

STATION NUCLEAR SAFETY AND OPERATING COMMITTEE REVIEW: 38

☒ APPROVED ☐ DISAPPROVED ☐ APPROVED AS MODIFIED

10/05/2001 12:15 2750 ENG

#888.10 B

DESIGN CHANGE REQUE
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

REMARKS: 39	DESIGN CHANGE NO.: 78-525
CHAIRMAN'S SIGNATURE: [signature] 41	DATE: 42 2/7/79 43

NUCLEAR ENGR. SERVICES' REVIEW:
ORGANIZATION TO CONDUCT REVIEW ~~OR FINAL DESIGN~~:
☒ NUCLEAR ENGR. SERVICES STAFF ☐ CONTRACTOR ☐ OTHER

PROJECT ENGINEER: S.W. Bristow, Jr. 44	DATE ASSIGNED: 2-6-79 45
AFFILIATION: Engineer - NES 46	

NUCLEAR ENGR. SERVICES' REVIEW: 47
UNREVIEWED SAFETY QUESTION ☒ NO ☐ YES COMMENT:

SUPERVISOR NUCLEAR ENGR. SERVICES' SIGNATURE: [signature] 48	DATE: 3/6/79 49

SYSTEM NUCLEAR SAFETY AND OPERATING COMMITTEE REVIEW: 50
UNREVIEWED SAFETY QUESTION: ☐ YES ☒ NO
☒ APPROVED ☐ DISAPPROVED ☐ APPROVED AS MODIFIED

COMMENTS: 51

CHAIRMAN'S SIGNATURE: [signature] 52	DATE: 3/12/79 53
FINAL DESIGN COMPLETED: L. LOBO 54	DATE: 3-30-79 55
TITLE: ASSISTANT ENGINEER 56	AFFILIATION: VEPCO 57
FINAL DESIGN REVIEWED BY STATION NUCLEAR SAFETY AND OPERATING COMMITTEE: 58 CHAIRMAN'S SIGNATURE: J. L. WILSON	DATE: 59 APR 9 1980
FINAL DESIGN IMPLEMENTATION CONTROLLING AND TESTING PROCEDURES 60 COMPLETED BY: L. LOBO	DATE: 61 3-20-79
REVIEWED BY STATION NUCLEAR SAFETY AND OPERATING COMMITTEE: 62 CHAIRMAN'S SIGNATURE: J. L. WILSON	DATE: 63 4-9-80
DATE DESIGN CHANGE COMPLETED ON UNIT NO. 1: 64 9-22 80	DATE DESIGN CHANGE COMPLETED ON UNIT NO. 2: 65 6-21 80
CONTROLLED DOCUMENT REVIEW AND REVISION COMPLETED BY PROJECT ENGINEER: Lawdman Lobo 66	DATE: 67 3-29-82
COMPLETED DESIGN CHANGE REVIEWED BY DESIGN CONTROL ENGINEER: [signature] 68	DATE: 69 3-29-82
COMPLETED DESIGN CHANGE AUDITED BY QUALITY ASSURANCE ENGINEER: [signature] 70	DATE: 71 4-1-82

FORWARD TO STATION RECORDS

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880.11 A

ENGINEERING REVIEW
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

ATTACH TO: DESIGN CHANGE REQUEST 1	DESIGN CHANGE NO: 78-S25 2
DESIGN CHANGE TITLE: Removal of Part Length Control Rods 3	
PROJECT ENGINEER PERFORMING REVIEW: Lawrence Lobo — Lawrence Lobo 4	DATE: 11/27/78 5
REVIEWED BY DESIGN CONTROL ENGINEER: R. H. Coupe — [signature] 6	DATE: 11/29/78 7
REVIEWED BY SUPERVISOR – ENGINEERING SERVICES: T. A. Peebles — T A Peebles 8	DATE: 11/30/78 9

ENGINEERING REVIEW: (THE REVIEW SHALL CONSIST OF: (1) ANALYSIS OF THE REQUEST: (2) PROPOSED RESOLUTION: (3) APPROVAL LEVEL:) 10

(1) ANALYSIS OF THE REQUEST:

1.) This design change request consists of the removal of part length control rods from Surry #1 and #2 Units. There are five part length control rod assemblies in each unit. After removing the part length control rods from the core, thimble plugs are to be inserted in the fuel assembly from which the part length rods are removed.

The intent of the part length control rods was to control axial power distribution and to suppress Xenon oscillations.

The utilization of part length control rod for axial power distribution control is not desirable. The insertion of the part length control rods would cause the lowering of power in the axial region surrounding neutron absorbing material. At the same time causing a higher power in the axial region just below and above the neutron absorbing material of the part length rod.

At the time Surry units were designed, there was no stringent restriction on $\Delta\phi$ band. At the present time, there is a restriction on maintaining a narrow $\Delta\phi$ band of ± 5% which reduces the Xenon oscillations to a very low level.

2.) Westinghouse has evaluated and analyzed the removal of the part length control rods while leaving the lead screw in the fully withdrawn position (Details are discussed by Westinghouse in a letter to B. R. Sylvia) and found:

(1) There are no thermal or hydraulic problems including no change in T_H in the upper head provided the part length rods are replaced by thimble plugs.

(2) There are no problems with replacing the part length rod with a thimble plug.

(3) There are no mechanical problems including vibrations, provided the lead screw is adequately supported at the top end. This can be done using an Anti-rotation Device. When the part length rod is unlatched, the lead screw is free to rotate. So when the screw is moved to the top of its housing, its own weight and/or vibration can cause it to rotate in the direction which would lower it. Westinghouse has designed a 40 year anti-rotation device that can be utilized to prevent the lead screw from rotating. The device has a pin which fits into holes drilled into both the anti-rotation device housing and the cap of the conoseal. The cap cannot rotate

4 899.12

ENGINEERING REVIEW (SUPPL: :NT)
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

ATTACH TO: ENGINEERING REVIEW | DESIGN CHANGE NO. 78-S-25

ENGINEERING REVIEW (CONTINUED):

(1) ANALYSIS OF THE REQUEST: (CONTINUED)

therefore the device cannot. The anti-rotation device can be installed while the head is in its laydown area.

The removal of the part length control rods provides the following benefits.

(1) Decreased outage time.

The design of the part length control rod drive mechanism is such that the lead screw, which is used to raise and lower the rod, cannot be removed from the mechanism. This results in the requirement for a removable seal at the top of the part length control rod drive mechanism, as well as a long tool for extending down into CRDM to unlatch the screw from the part length rod. This unlatching and relatching process can require as much as two 10-hour shifts during each refueling outage, all of which can be critical path time. Removal of the part length control rods can therefore save as much as a full day of outage time.

In addition, after the part length control rods are removed, the seals at the top of part length lead screw travel housing need never be opened during a refueling. Because the seal is never broken, this virtually eliminates any possibility of leakage during plant startup, following an outage. Therefore, the risk of significantly extending the outage while cooling down, depressurizing, and repairing a leak at this location, is reduced essentially to zero.

(2) Decreased radiation exposure

The latching/unlatching process requires two individuals at a time working for as much as a total of 20 hours in a high radiation field. After the part length rods are removed, none of this is necessary. This makes a significant contribution to the ALARA program.

(2) PROPOSED RESOLUTION:

Based on the Westinghouse study, and operational experience at Surry, it is recommended that the following be accomplished: (1) Remove part length control rods from the core, (2) Insert thimble plugs in the fuel assemblies, which contain part length control rods, (3) Install Anti-rotation Device to keep the lead screw in the raised position.

During the fuel shuffle, the part length control rods may be inserted into spent fuel assemblies and taken to the spent fuel pit while thimble plugs are inserted into the locations formerly occupied by the Part Length Control Rods.

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1. GENERAL INFORMATION.

1.1 General data.

1.1.1 General description.

The 157-inch reactor vessel consists of a vessel shell and a closure head. The vessel shell is a cylindrical section with a 12-feet 11-7/8-inch I.D. and a 14-feet 5-7/32-inch O.D. at the primary inlet and outlet connections. Below these connections it has a 13-feet 1-5/16-inch I.D. and a 14-feet 5-7/16-inch O.D.
The dimension from the centerline of the vessel to the outer face of the inlet nozzle is 10-feet 5-½-inches. The dimension from the centerline of the vessel to the outer face of the outlet nozzle is 10 feet 2-3/8 inches.
The bottom hemispherical head is machined to receive 50 instrumentation nozzles. The closure head is machined to receive the 65 control rod mechanism housings.
The vessel stands 42 feet 7-3/16 inches high from the bottom hemispherical head to the top of the control rod mechanism housings. (see also figure 1-1).

1.1.2 Design conditions.

Design pressure	2485 psig
Design temperature	650° F.
Hydrotest pressure	3107 psig
Hydrotest temperature	NDTT + 60° F minimum
Hydrotest temperature at manufacture	110° F

1.1.3 Operating conditions.

Coolant fluid	Pressurized water
Operating pressure	2235 psig
Normal operating temperature	543° F
Inlet temperature	543° F
Outlet temperature at normal power	605,8° F

ONLY THIS INFORMATION IS PERTINENT ON THIS PAGE.
WJDeCoo

1.1.4 Initial operating limitations.

The heating and cooling rate is limited to maximum 100°F per hour. These rates will be safe for 200 occurances each. Thus, when starting at an isothermal condition at 100°F, the maximum heating rate is not to exceed 100°F per hour up to operating temperature and, when starting at an isothermal condition at operating temeperature, the maximum cooling rate is not to exceed 100° F per hour returning to 100° F.

1.1.5 Basic Dimensions.

1.1.5.1 Vessel Shell Assembly.

Flange Forging	15-feet 4-inch O.D. x 2-feet 11-½ inch length
Cylindrical Section Nozzles	12-feet 11-7/8-inch I.D. x 9-inch minimum thick manganese-molybdenum steel plus 0.156-inch austenitic stainless steel cladding.

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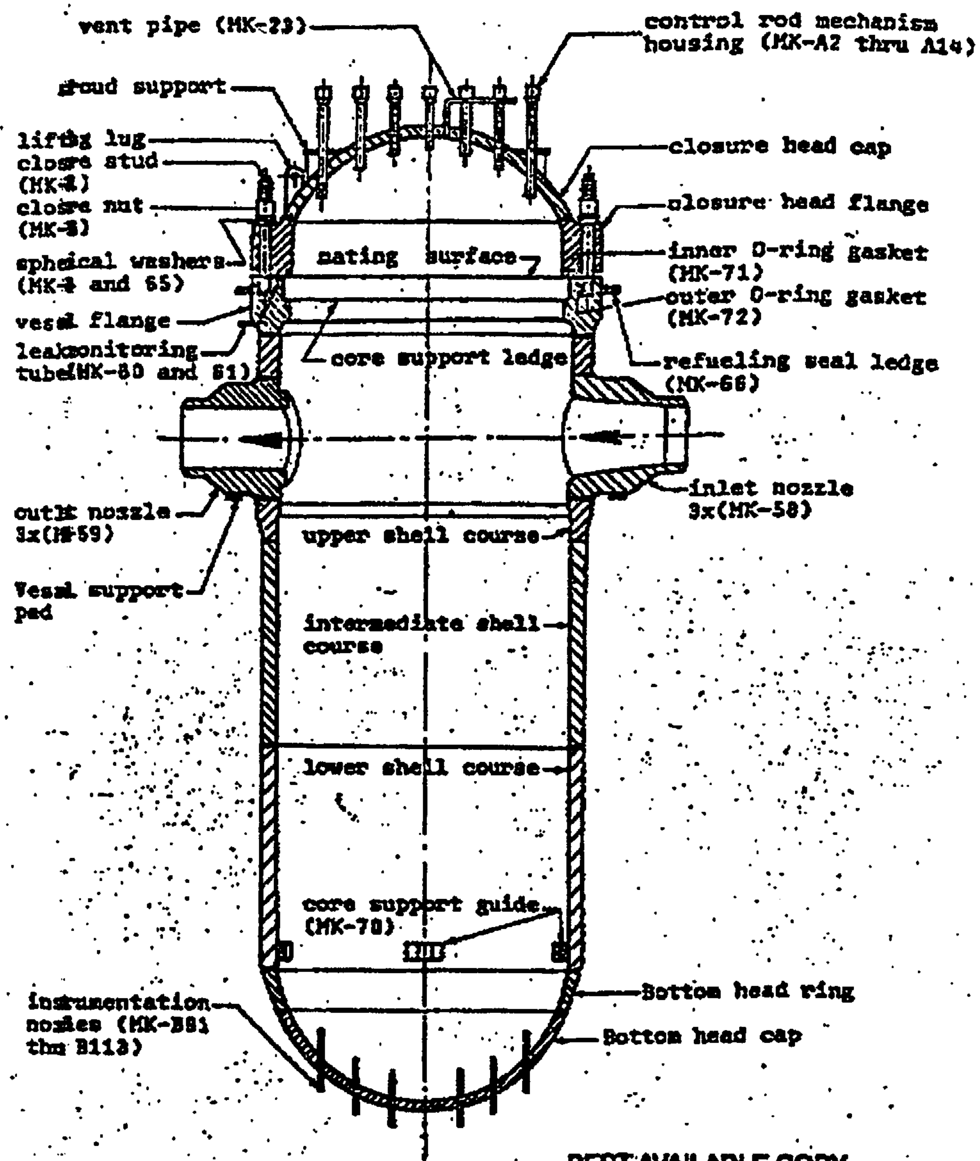

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Figure 1-1 Reactor vessel

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Cylindrical Section	13-feet 1-5/16 inch I.D. x 8- inch minimum thick manganese-molybdenum steel plus 1/8-inch austenitic stainless steel cladding.
Hemispherical Head	6-feet 7-½ inch spherical radius x 5 inch minimum thick manganese-molybdenum steel plus 1/8-inch austenitic stainless steel cladding

1.1.5.2. <u>Closure Head Assembly.</u>

ONLY THIS INFORMATION IS PERTINENT
W.J. Le Coure

Closure Head Forging	15-feet 4-inch O.D. x 2-feet 11-11/32 inch length.
Closure Head Plate	6-feet 7-½ inch spherical radius x 6-3/16 inch minimum thick manganese-molybdenum steel plus 1/8-inch austenitic stainless steel cladding.
Studs	6 inch nominal diameter x 5-feet length.

1.1.6 <u>General Dimensions.</u>

Overall Height of Reactor Vessel Assembly	
Including Control Rod Housings	42 feet 7-13/64 inches
Excluding Control Rod Housings and Instrumentation Nozzles	40 feet 6- 3/32 inches
Overall Height of Reactor Vessel Excluding Closure Assembly and Instrumentation Nozzles	33 feet 10-49/64 inches
Outside Dimension from Centerline of Shell to Face of Outlet Nozzles	10 feet 2-3/8 inches
Outside Dimension from Centerline of Shell to Face of Inlet Nozzles	10 feet 5-½ inches
Outside Diameter of Shell at Nozzles	174-7/32 inches
Outside Diameter of Shell Below Nozzle Section	173-7/16 inches
Outside Diameter of Refueling Seal Ledge	197.000 inches
Outside Dimension from Centerline of Shell to Lifting Lugs	6 feet 3-½ inches
Dimension from Centerline of Shell to Lifting Lug Hole Centerline	5 feet 11 inches
Shell Thickness Including Cladding:	
Flange, Maximum (Pressure Boundary)	1- foot 5-7/32 inches
Flange, Minimum (Pressure Boundary)	1- foot 5-3/16 inches
Upper Shell Course, Minimum	9- 1/8 inches
Intermediate Shell Course, Minimum	8- inches
Lower Shell Course, Minimum	8- inches
Lower Head Ring, Minimum	5- 1/8 inches
Bottom Hemispherical Head, Minimum	5- 1/8 inches
Hemispherical Closure Head, Minimum	6-5/16 inches

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1.1.7. Dry Weights.

Reactor vessel	559,082	lb.
Reactor Closure Head	111,347	lb.
Studs, Nuts & Washers	31,563	lb.
Total Assembled Reactor Vessel Weight	701,992	lb.
Closure Stud Assembly		
Stud (MK-62) (Includes Inserts MK-78 & MK-79)	450.38	Lb. each
Nut (MK-63)	64.7	lb. each
Spherical Washer Set (MK-64 & MK-85)	29	lb. each
Total per Set	544.18	lb. each set
Total for 58 sets	31,563	lb.
Vessel Shipping Arrangement		
Reactor Vessel	559,082	lb.
Roll-on/Roll-off skid	26,455	lb.
Miscellaneous Shipping parts	6,614	lb.
Total Reactor Vessel Shipping Weight	592,151	lb.
Closure Head Shipping Arrangement		
Closure Head	111,347	lb.
Shipping Skid and Cover	7,496	lb.
Mechanism Housing Cover	3,527	lb.
Total Closure Head Shipping Weight	122,370	lb.

ONLY THIS INFORMATION IS PERTINENT.
W.J. De Coom

1.1.8. Design Considerations.

The materials produced and used in fabrication of the reactor vessel (under this contract) are in accordance with the qualifications indentified in Paragraphs 1.1.8.1 and 1.1.8.2.

1...8.1. Governing Specifications.

1. A.S.M.E. - Code Section III.

2. A.S.M.E. - Code Section IX.

3. Westinghouse P.W.R. Equipment Specification 676413 and 677026.

1.1.8.2. Material Specifications.

The material specification for each Mark Number is listed in Figure 7.26.

1.1.9. Safety Notices and Warnings.

The internal surfaces of the reactor vessel come in contact with radioactive primary coolant of the neclear power plant; therefore, radioactive materials will be present during operatic and may be present for long periods after shutdown. Personnel working at or near this vessel should be thoroughly familiar with the hazards involved.

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The reactor vessel is designed to operate at temperatures up to 650°F and fluid pressures up to 2485 psig. It has a high hydrostatic test pressure (3107 psig). Due regard must be made for these conditions to minimize the danger of injury to personnel.

ONLY THIS INFORMATION IS PERTINENT
W.J. [illegible]

The minimum temperature for pressurization is NDTT +60°F (110°F minimum) at time of manufacture.

The reactor vessel shell is fabricated of ASTM A-508, Class 2, manganese-molybdenum steel. Since this material has a high brittle fracture transition temperature, extreme care must be taken by all persons working on and/or handling this equipment. No welding, striking of arcs, notches, grooves, or other stress concentrations shall be allowed on the surface of the vessel at any time during handling, installation, or operation. In the event such an incident occurs the matter shall be immediately reported to the Plant Operations Engineer. No remedial action shall be initiated except as directed by the Plant Operations Engineer.

1.2. Installation and Maintenance Operations.

1.2.1. Cleaning.

WARNING

Improper mechanical or chemical cleaning of surfaces may result in excessive local corrosion of those surfaces when placed in contact with primary coolant. The resultant corrosion products taken into solution in the primary coolant could become highly radioactive, thus complicating the maintenance of any component due to the hazards of exposing men to high levels of radioactivity.

CAUTION

Use extreme care at all times to prevent dirt, foreign particles, etc., from entering the reactor system and lodging between bearing surfaces of parts operating with extremely small clearances and causing excessive wear or seizure.

NOTES

1. Components shall be cleaned to the extent that no contamination is visible. Areas which cannot be visually inspected due to inaccessibility or geometry shall be evaluated by wiping the surface with a wet or dry, lint-free cloth until all traces of foreign material are removed and the cloth remains clean after use.

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2. Rust of any type or amount shall not be allowed. If rusting does occur, the surface shall be cleaned to remove the rust or rust-producing condition and any visible surface contamination.

3. Cleanliness shall be maintained by packaging components or subassemblies in polyethylene bags for storage.

All instructions for the cleaning of surfaces in this instruction manual refer to a condition of maximum cleanliness. The cleaning is to be performed as follows :

1. Clean all metal surfaces as necessary by swabbing with clean, lint-free cloths saturated with acetone followed by swabbing with clean, lint-free cloths saturated with destilled water. Dry with clean, lint-free cloths. The cleaning must be such that no foreign matter can be seen after cleaning, particulary in the root area of the threads.

2. Clean Buna-N Rubber as necessary by swabbing with clean, lint-free cloths saturated with chloride-free naphta gas followed by swabbing with clean, lint-free cloths saturated with destilled water. Dry with clean, lint-free cloths. The cleaning must be such that no foreign matter can be seen after cleaning.

3. Pressure sensitive tape may be used occasionally on components (that is, over the top of closure studs). Any time the pressure sensitive tape is removed from a component, use acetone to remove any residue. Clean the area as described above in Step 1.

THE INFORMATION OF THIS PAGE IS NON-PERTINENT
WJDeC

1.2.2. Lubrication.

As the following tabulated parts are assembled, they shall be lubricated as indicated below.

Mark No.	Nomenclature	Lubricant	Apply to
MK-62	Stud	Neolube	Male threads
MK-63	Nut	Neolube	Bearing surface
MK-64	Convex Spherical Washer	Neolube	Both faces
MK-65	Concave Spherical Washer	Neolube	Both faces
MK-78	Top Insert	Neolube	Male threads
MK-79	Bottom Insert	Neolube	Male threads
MK-80	Eyebolt	Neolube	Male threads
	Plug (Westinghouse)	Neolube	Male threads
MK-32	Sleeve	Neolube	Male threads
MK-28	Guide Stud	Neolube	Bottom 8-inches
MK-31	Eyebolt	Neolube	Male threads

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3 DESCRIPTION.

3.1 Detailed Description.

(See figures 7.1, 7.8, 7.9, 7.10, 7.11, 7.15, 7.20 and 7.26)

3.1.1 Introduction.

The Virginia Electric and Power Company reactor pressure vessel equipment described in this manual include: the vessel, the closure head assembly, closure stud assembly, special tools, and shipping arrangements. Discussions of the equipment with detailed description of their features are presented in subsequent paragraphs. Material and material specifications for all parts or segments are presented in Figure 7.26 by mark numbers.

3.1.2 Vessel Shell Assembly.

The reactor vessel (see figures 7.11, 7.2, 7.3, 7.4 and 7.5) is built up from :

(1) A flange forging.

(2) A refueling seal ledge.

(3) An upper shell course containing the inlet and outlet nozzles.

(4) An intermediate shell course.

(5) A lower shell course containing the core support guides.

(6) A lower head ring.

(7) A bottom hemispherical head having the instrumentation nozzles.

The vessel segments are discussed in subsequent paragraphs.

3.1.2.1 Reactor Vessel Flange.

The reactor vessel flange is a machined forging welded to the upper shell course. (See figure 7.3).
A refueling seal ledge is welded to the vessel flange. The flange is fabricated of ASTM A-508, Class 2, manganese-molybdenum steel and is clad internally and on the gasket face with weld deposited austenitic stainless steel.

The flange is designed with a ledge for the support of the core, a gasket face for sealing of the vessel, 2 monitoring taps on 95°33' and 139°27' degrees angular location for detection of water leakage through the gasket closure, irradiatio tube slots on 45°, 55°, 65°, 165°, 245°, 285°, 295°, 305° degrees angular location for holding of irradiation specimen baskets, key slots on 0, 90, 180 and 270 degrees angular location for aligning the closure head and vessel assembly and 58 stud holes for tightening the head to the vessel.

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Of these stud holes 3 holes are used for holding the guide studs which are used for refueling. The stud holes are threaded and receive the 6 inch diameter closure studs.

3.1.2.2. Refueling Seal Ring Ledge.

The refueling seal ring ledge (See figure 7.5) is a machined weldment fabricated of ASME SA-533, Grade A, manganese-molybdenum steel. The refueling seal ledge is a 2-½-inch thick ring welded to the reactor vessel flange.

3.1.2.3. Upper Shell Course.

The upper shell course of the vessel (see figure 7.3) is a machined forging welded to the reactor vessel flange and to the intermediate shell course. The upper shell course is fabricated of ASTM A-508, Class 2, manganese-molybdenum steel and is clad internally with weld deposited-stainless steel. The upper shell course contains the six primary coolant nozzles.

The six primary coolant nozzle forgings are welded to the upper shell course for entry and discharge of the primary coolant. The nozzle centerlines are 6 feet 10-7/16 inches below the mating surface of the vessel flange.

The three 27.469-inch I.D. inlet nozzles are located 120 degrees apart, (their centerlines are located respectively on 95, 215 and 335 degrees).

The three 28.969-inch I.D. outlet nozzles are located 120 degrees apart (their centerlines are located respectively 25, 145 and 265 degrees). Vessel support weld pads are located on the bottom of each of the six nozzles. The machined pads are 9 feet 2-15/16 inches below the mating surface of the vessel flange.

The primary coolant nozzle forgings are also fabricated of ASTM A-508, Class 2, manganese-molybdenum steel and are clad with weld deposited austenitic stainless steel internally. The nozzle and connections are clad with weld deposited austenitic stainless steel and are machined for field welding to the main coolant piping.

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3.1.2.4. Intermediate Shell Course.

The intermediate shell course (see figure 7.2) is a cylindrical shell formed from two plates of ASTM A-533 Gr.B Cl.1, manganese-molybdenum steel and is clad internally with weld deposited austenitic stainless steel. The intermediate shell course is welded to the upper and lower shell courses. The two longitudinal weld seams are located on 45 and 225 degrees.

3.1.2.5. Lower Shell Course.

The lower shell course (see fig.7.2) is a cylindrical shell formed from two plates of ASTM A-533 Gr.B Cl1, manganese-molybdenum steel and is clad internally with weld deposited austenitic stainless steel except for the weld deposited Inconel cladding on the bottom 11-3/16 inches. Four core support guides which have a 8-1/16 inch wide x 4.040 inch deep x 8-½ inch long machined slot at the bottom of the shell course are located on 0, 90, 180 and 270 degrees. The core support guides are fabricated of ASME SB-166-63 Inconel.

The lower shell course is welded to the intermediate shell course and to the lower head ring.

The two longitudinal weld seams are located on 135 and 215 degrees.

3.1.2.6. Lower Head Ring.

The lower head ring (see figure 7.2) is welded to and joins the lower shell course and the bottom hemispherical head. It is fabricated of ASTM A-508, Class 2, manganese-molybdenum steel and is clad internally with weld deposited austenitic stainless steel.

3.1.2.7. Bottom Hemispherical Head.

The bottom hemispherical head (see figures 7.1 and 7.2) is welded to the lower head ring of the vessel The hemispherical head is formed from a single plate of ASTM A-533, manganese-molybdenum steel and is internally clad with 0.125-inch thick weld deposited austenitic stainless steel. The head is penetrated by 50 instrumentation nozzles fabricated from ASME SB-166-63 Inconel.

Each 1-½ inch O.D. (0.607 inch I.D.) instrumentation nozzle is Inconel welded into place. A safe end of ASME SA-475, Type 304, stainless steel is welded to the exterior end of each instrumentation nozzle.

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3.1.3. Closure Head Assembly.

The closure head assembly (see figures 7.6, 7.7, 7.8, 7.9, 7.10 and 7.12) is a weldment consisting of a hemispherically dished plate and a flange forging. The hemisph. dished plate is fabricated of ASTM A-533 Gr.B Cl.1, manganese-molybdenum steel and is clad internally with weld deposited austenitic stainless steel 0.125 inch thick.

The flange forging is ASTM A-508, Class 2, manganese-molybdenum steel and is clad with weld deposited austenitic stainless steel internally and on the gasket face. The closure head forging gasket face is machined to accommodate two silver plated self-energizing stainless steel O-ring gaskets and the 24 sets of wire clips, backing plates, and screws.
The flange of the forging is bored through to receive the 58 closure head studs. An indicator arrow is welded to the head to indicate the number one stud hole.

The dished segment of the closure head contains 65 penetrations, positioned in a square pattern on 8.466 inch centers, to accommodate the control rod mechanism housings. A nominal one-inch diameter penetration in the closure head accommodates the vent pipe.

ONLY THIS INFORMATION IS PERTINENT.

The closure head has three lifting lugs. Three vent shroud support lugs are also attached to the closure head.

3.1.3.1. Control Rod Mechanism Housings.

Each of the 65 control rod mechanism housings (see figure 7.13) penetrating the closure head is a weldment consisting of a threaded, 6-inch O.D. adapter and a 4-inch O.D. body. The adapter is fabricated of ASME SA-182, Type 304, stainless steel, and the body is fabricated of ASME SB-167 Inconel.

The mechanism housing weldments are inserted with an interference fit into the penetrations of the closure head. The bodies are welded into the inside of the closure head with weld deposited Inconel.

3.1.3.2. Vent Shroud Support Assembly.

The vent shroud support assembly (see figure 7.9) is attached to the closure head at three places. Each pair of support lugs on the vent support ring is mated with a vent shroud support lug on the closure head assembly and is fastened to it by a 3/4-inch hex head bolt with nut.

The shroud support flange has 18 holes of 11/16-inch diameter, equally spaced on a 128-inch diameter bolt circle. The flange is welded to the support ring; and the assembly is stiffened by 15 support gussets welded to the ring and flange at equal distances.

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The 24 shroud insulation support angles are equally spaced on and welded to the support ring. In addition, the support ring has 24 saw cuts, each terminating in a 1/2-inch diameter hole. The saw cuts and holes are equally spaced between the support angles. The saw cuts enable the support ring to compensate for temperature caused variations in dimensions; this will allow the support lug attachments to remain secure.

3.1.3.3. Closure Stud Assembly.

The closure head is secured to the vessel flange by 58 closure stud assemblies. (see figure 7.20) Each assembly consists of a threaded, hex head stud with a nominal 6-inch diameter, a nut having eight castellations at the top, a set of spherical washers, and top and bottom inserts.

Each stud has a one-inch diameter center hole through the length of the stud to receive a stud elongation measuring rod. The bottom insert is used to close the bottom of the stud and serves as a seat for the stud elongation measuring rod. The top insert is used to close the top of the stud and prevents the entry of any foreign matter. Each stud has a threaded length sufficient to accommodate a hydraulic stud tensioner. For handling purposes an eyebolt is supplied for each stud. The studs, nuts and spherical washers (marked in matched sets) are fabricated of ASTM A-540, Gr. B 24, nickel-chrome-molybdenum steel. The studs and washers are "phosphated".

3.1.4. Special Tools.

The special tools for mounting and measuring supplied by The Rotterdam Dockyard Company are listed in table 6.2. The identification and function of each tool are given in the table.

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WJDeCoom

APPENDIX D
Doc Id 51-5015197-00

Westinghouse Electric Company
Box 355
Pittsburgh Pennsylvania 15230-0355
Fax Number: (412) 374-6647

TO: DEAN PRICE

FROM: JUSTIN LEDGER **DATE:** 10/12/01

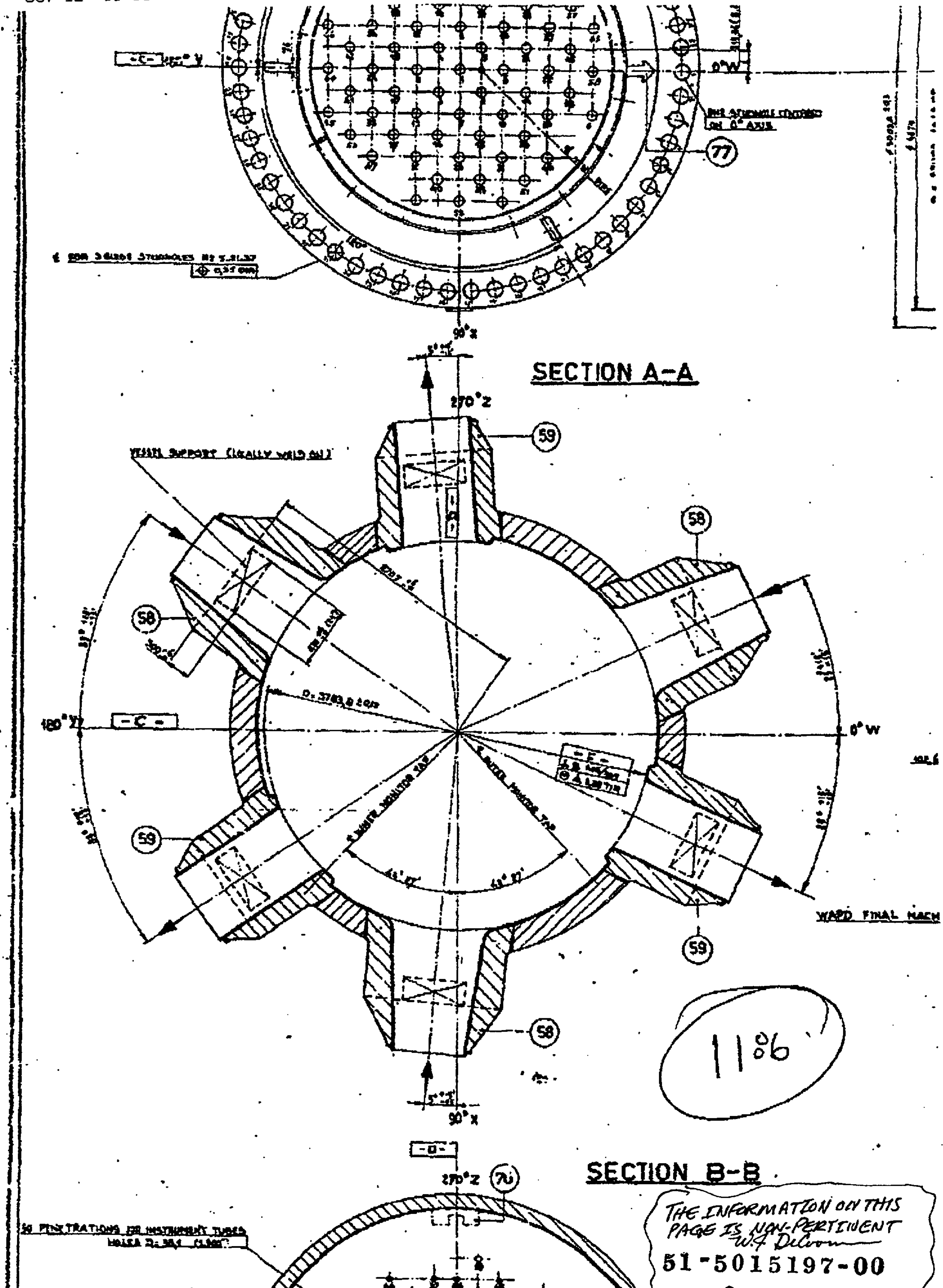
MESSAGE:

DEAN,

PLEASE FIND ATTACHED THE SECTIONS OF THE DRAWINGS YOU REQUESTED. IF YOU NEED FURTHER CLEARIFICATION, DO NOT HESITATE TO CALL ME,

(412) 374-3898

Number of pages 15 INCLUDING COVER



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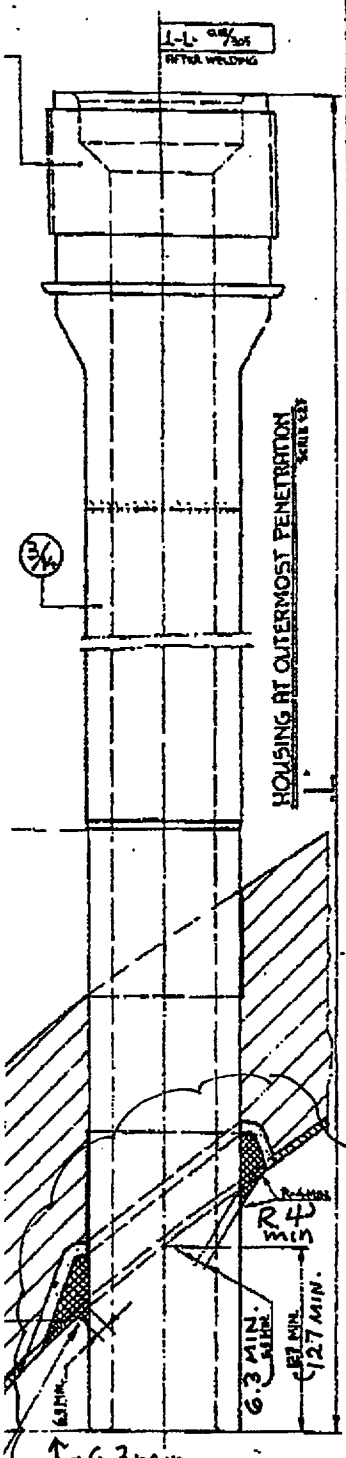
Pg 2



RADII SHOWN ON OPPOSITE SIDE MAY BLEND INTO A SINGLE 4mm. MIN. RADIUS ON THIS SIDE WHERE APPLICABLE.

ONLY THIS INFO IS PERTINENT, FOR LEGIBILITY.
WJ DeCooman

PENETRATION Nº:	HOUSING ASSY. Nº:	"L" (REF)	"G" (REF)
1	A2-137-1	965.20	328.61
2	A3-137-1	988.22	337.34
3	A3-137-2		
4	A3-137-3		
5	A3-137-4	988.22	337.34
6	A4-137-1	1012.03	342.90
7	A4-137-2		
8	A4-137-3		
9	A4-137-4	1012.03	342.90
10	A5-137-1	1059.66	351.63
11	A5-137-2		
12	A5-137-3		
13	A5-137-4	1059.66	351.63
14	A6-137-1	1084.26	355.60
15	A6-137-2		
16	A6-137-3		
17	A6-137-4		
18	A6-137-5		
19	A6-137-6		
20	A6-137-7		
21	A6-137-8	1084.26	355.60
22	A7-137-1	1158.08	366.71
23	A7-137-2		
24	A7-137-3		
25	A7-137-4	1158.08	366.71
26	A8-137-1	1185.07	371.48
27	A8-137-2		
28	A8-137-3		
29	A8-137-4	1185.07	371.48
30	A9-137-1	1210.47	375.44
31	A9-137-2		
32	A9-137-3		
33	A9-137-4		
34	A9-137-5		
35	A9-137-6		
36	A9-137-7		
37	A9-137-8	1210.47	375.44
38	A10-137-1	1291.43	388.94
39	A10-137-2		
40	A10-137-3		
41	A10-137-4		
42	A10-137-5		
43	A10-137-6		
44	A10-137-7		
45	A10-137-8	1291.43	388.94
46	A11-137-1	1375.57	402.43
47	A11-137-2		
48	A11-137-3		
49	A11-137-4	1375.57	402.43
51	A12-137-1	1404.94	407.99
53	A12-137-2	1404.94	407.99
55	A12-137-3	1404.94	407.99
57	A12-137-4	1404.94	407.99
58	A13-137-1	1435.10	413.54
59	A13-137-2		
60	A13-137-3		
61	A13-137-4	1435.10	413.54
62	A14-137-1	1496.22	423.86
63	A14-137-2		
64	A14-137-3		
65	A14-137-4		
66	A14-137-5		
67	A14-137-6		
68	A14-137-7		
69	A14-137-8	1496.22	423.86

LETTER	DATUM	OMSCHRIJVING VAN DE WIJZIGING
A	3-11-'69	CHANGED ACCORDING PAR 36, HOUSING ASSY. Nº CHANGED

AFGEGEVEN AAN	DATUM
W.E.C.	
LLOYDS	
AFD. BOUW	
SMEDERIJ	
MACH.FABR.	
COÖRD. BUR. W.	
LAB.	
LASSERS AFDV.	
MR. HERBERT	
RE: EXPED. K.E.	
K.E.	
MR. RUTHERFORD	
W4Z.	

1185 SHEET 1 OF 3

Note:

ALL CONTROL ROD, MECH. HOUSING CENTERLINES MUST BE PERPENDICULAR TO THE MATING SURFACE WITHIN 0.13mm./300mm.

REFERENCE DRAWINGS

CLOSURE HEAD SUB-ASSY.	30678 - 1184 (2 SHEETS)
CRD. HOUSING	30678 - 1188
GENERAL ARRANGEMENT	30678 - 1186

FOR SHROUD SUPPORT ASSY. SEE SHEET Nº2

FOR VENT PIPE ASSY. SEE SHEET Nº3

VPA-RCPCRV1

POS NR	AANT.	BENAMING	NORM.	MATERIAAL		BESTELNR.	GEW. IN KG.
49	4	RND HEAD MACH. SCREW		BRASS			
48	1	NAME PLATE		CAST BRONZE			
25	9	BOLT WITH NUT 3/4" x 76		B307			
3÷14	64	CRD. HOUSING BODY		INCONEL SB-167			
2	1	CRD. HOUSING BODY		INCONEL SB-167			
1	65	CRD. HOUSING ADAPTER		SA-182 TYPE 304			

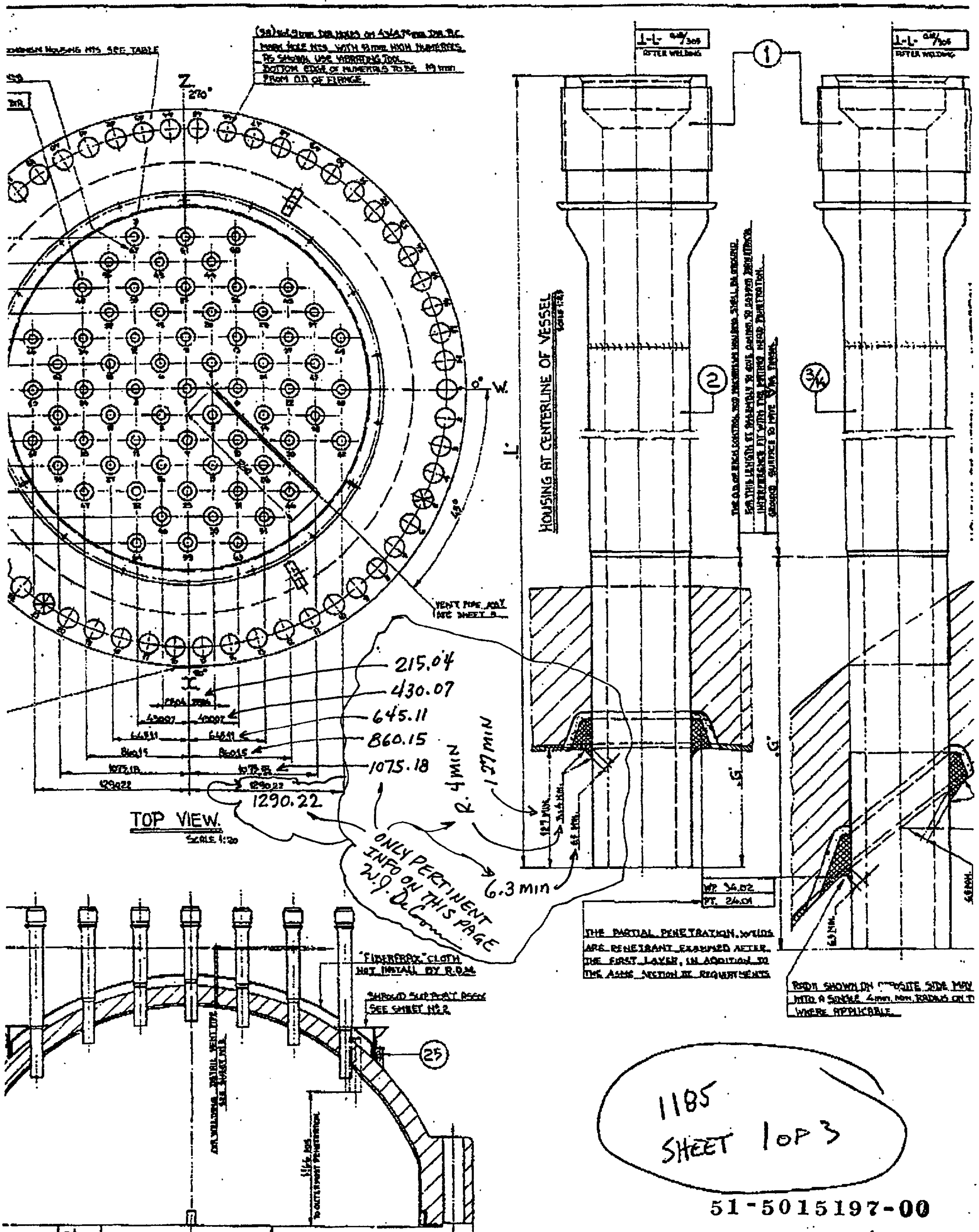
PAR.

STUKLIJST PER VOOR

DE ROTTERDAMSCHE DROOGDOK MIJ. N.V.

AFD. KERNENERGIE

"157" DWR VESSEL WESTINGHOUSE"



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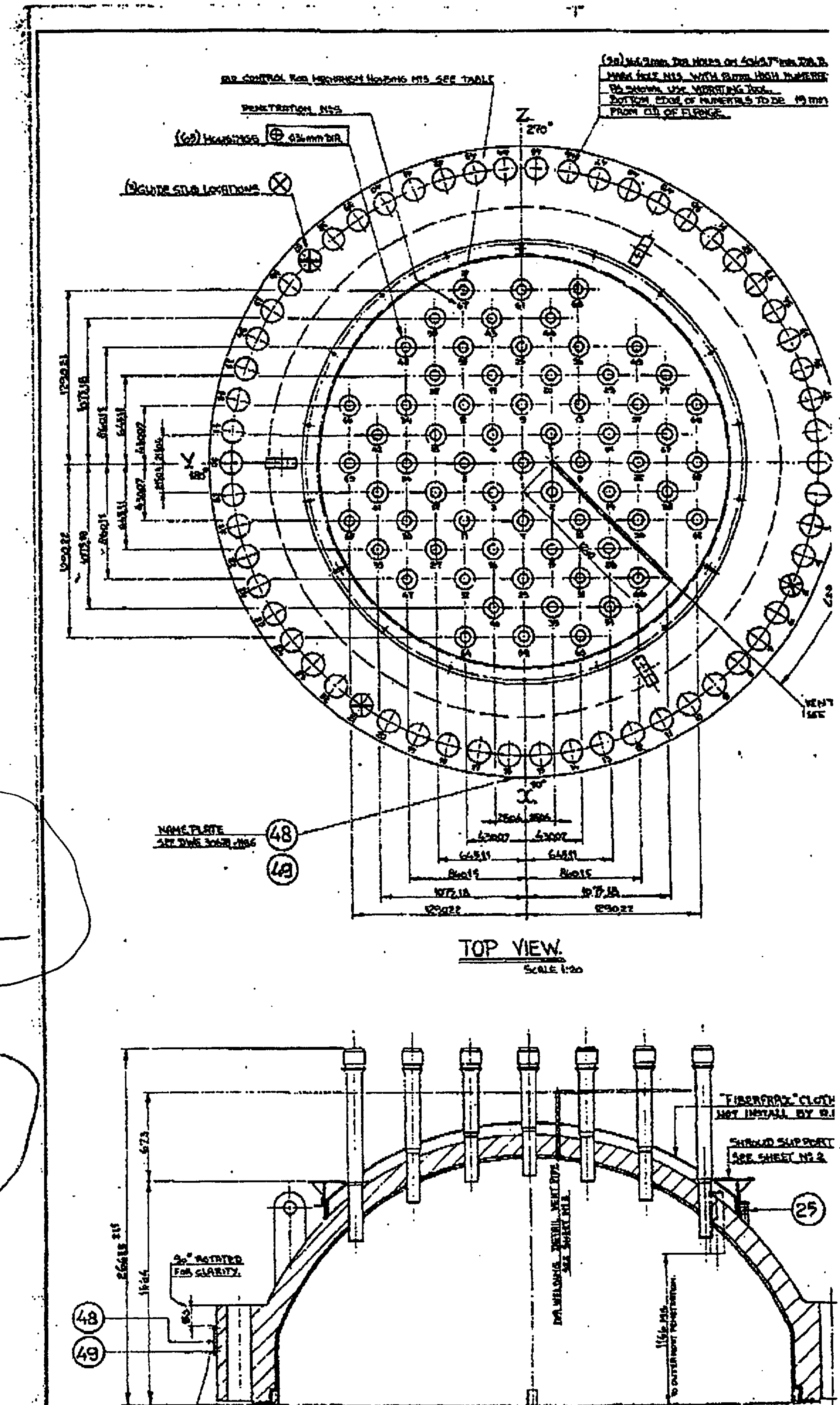

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W.J. DeCoon

1185
SHEET 1 OF 3

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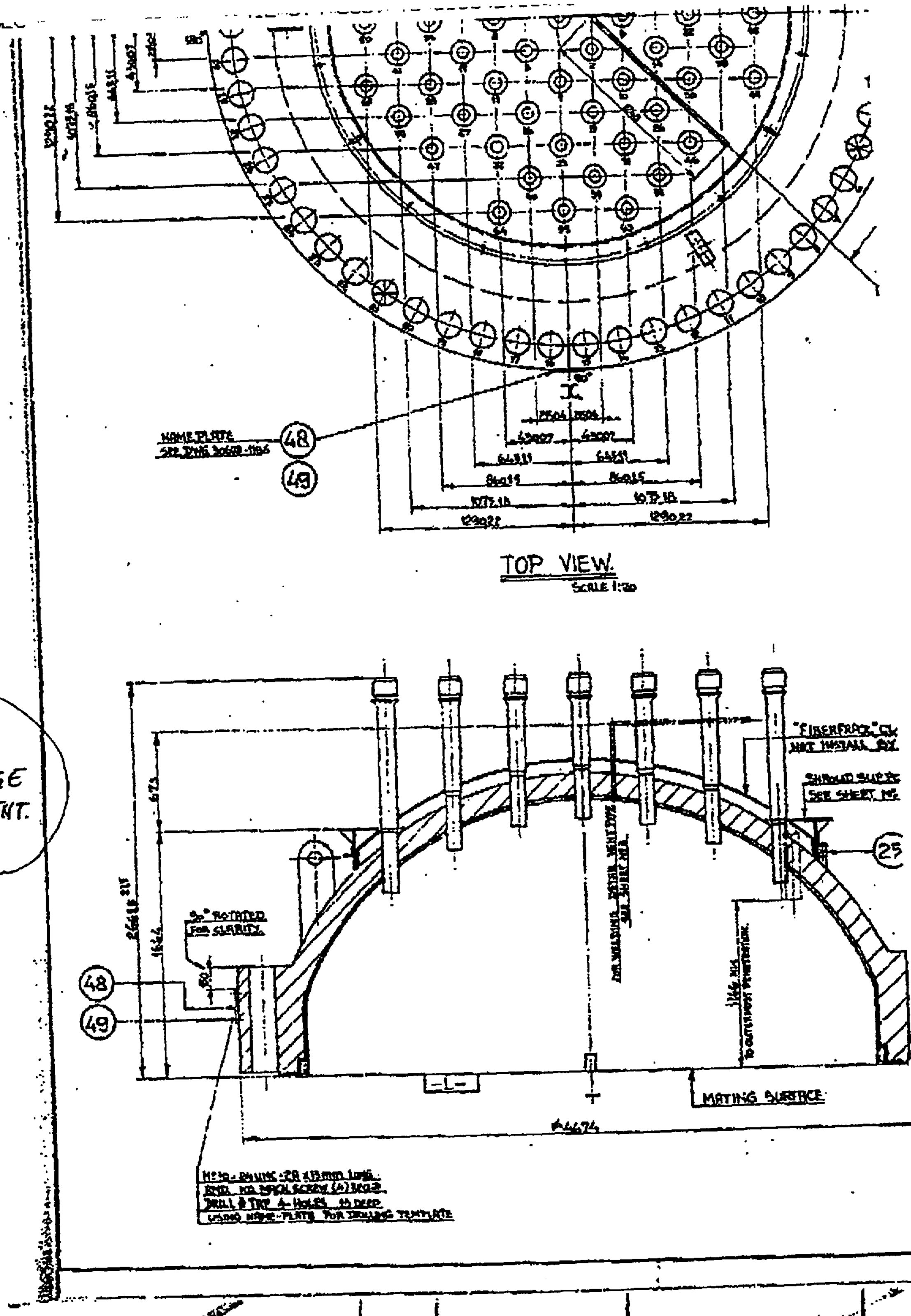

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SHEET 1 OF 3

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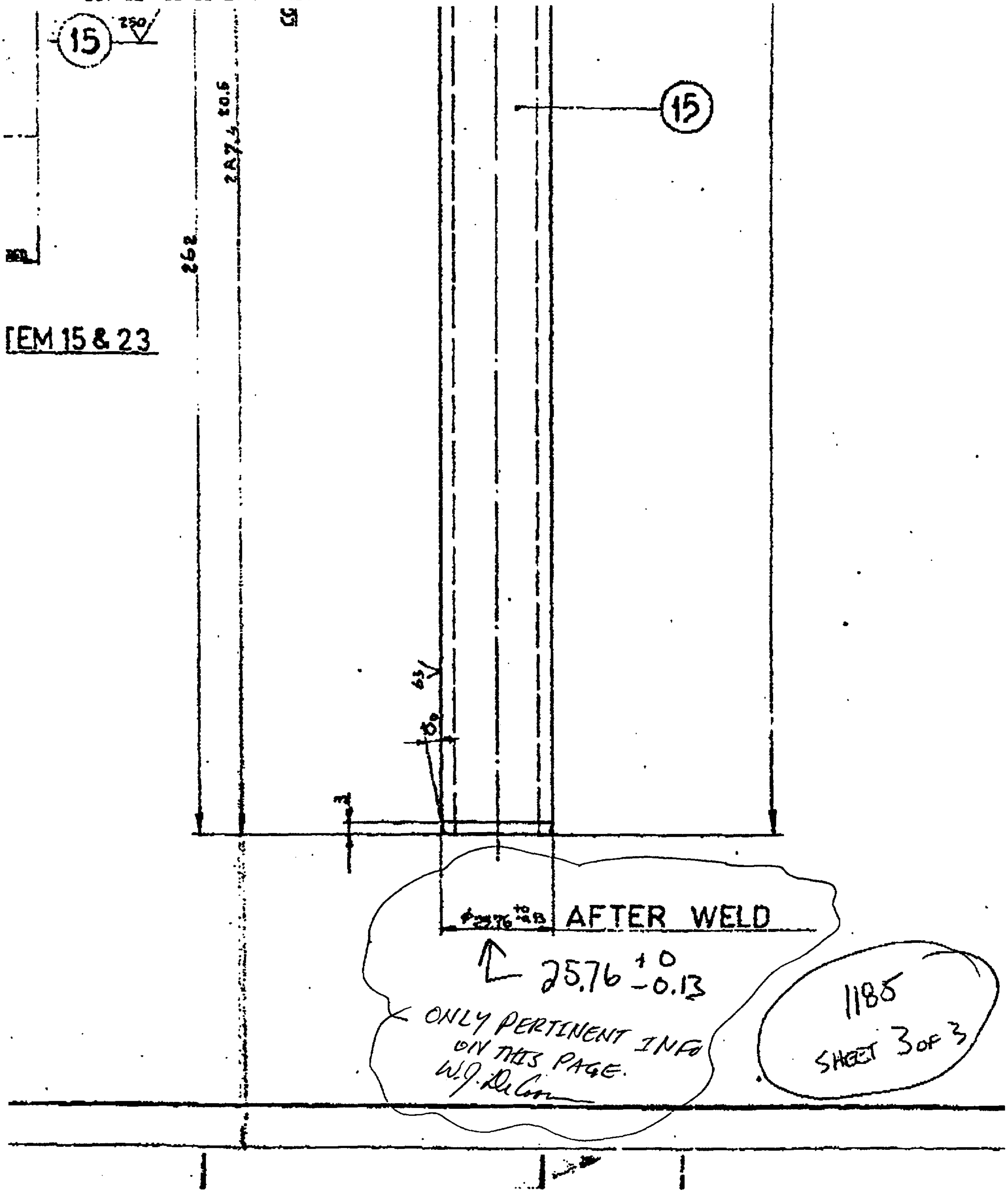
Pg. 6



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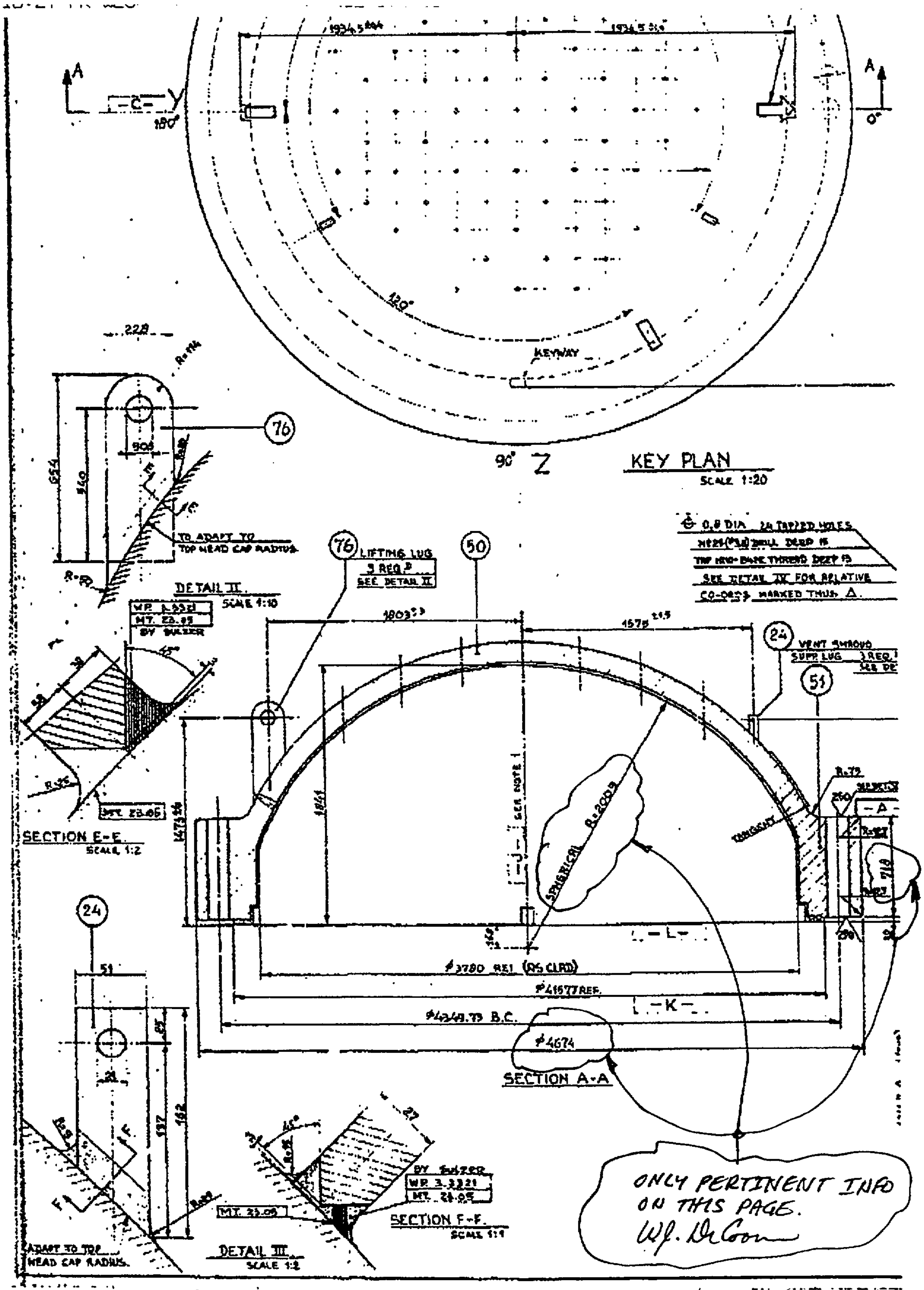
Pg. 7


KEY PLAN
SCALE 1:20
DETAIL II.
SCALE 1:10
SECTION E-E.
SCALE 1:2
SECTION A-A
DETAIL III
SCALE 1:2
SECTION F-F.
SCALE 1:1
LIFTING LUG
3 REQ'D
SEE DETAIL II
KEYWAY
TO ADAPT TO TOP HEAD CAP RADIUS.
ADAPT TO TOP HEAD CAP RADIUS.
SPHERICAL R.2000
TANGENT
BY SULZER
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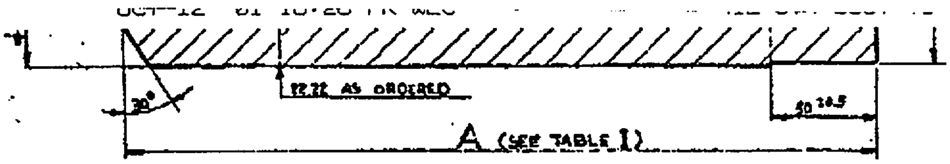
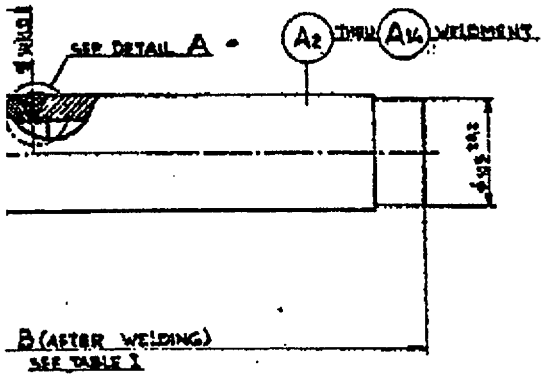
Wj. DeCoon

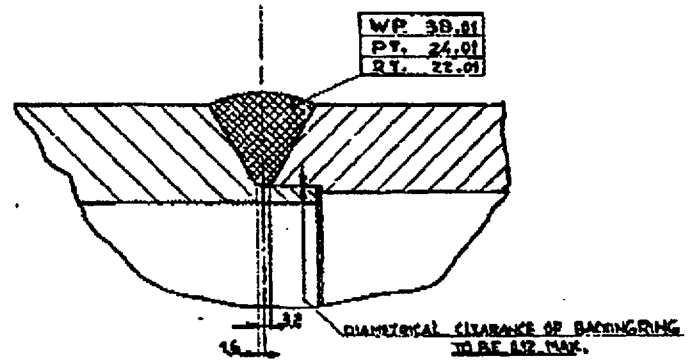
1184 FIG 7-7



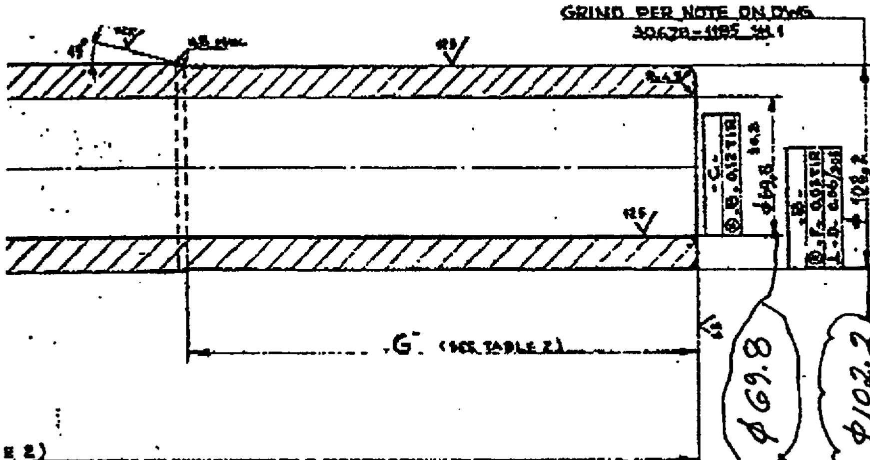
A10	1	10	1018	1304	[illegible]
A11	1	11	1074	1380	[illegible]
A12	1	12	1125	1418	[illegible]
A13	1	13	1153	1448	[illegible]
A14	1	14	1220	1509	[illegible]



DETAIL A



ROL ROD MECHANISM HOUSING



ϕ69.8

ϕ102.2

TABLE II				
HOUSING ASSY No	PENETRATION No	L	G	No REQ'D
A2	1	965	329	1
A3	2,3,4,5	988	337	4
A4	6,7,8,9	1012	343	4
A5	10,11,12,13	1059	352	4
A6	14,15,16,17 18,19,20,21	1084	356	8
A7	22,23,24,25	1158	367	4
A8	26,27,28,29	1185	372	4
A9	30,31,32,33 34,35,36,37	1210	375	8
A10	38,39,40,41 42,43,44,45	1291	389	8
A11	46,47,48,49	1375	402	4
A12	51,53,55,57	1465	408	4
A13	58,59,60,61	1435	414	4
A14	62,63,64,65 66,67,68,69	1486	414	8

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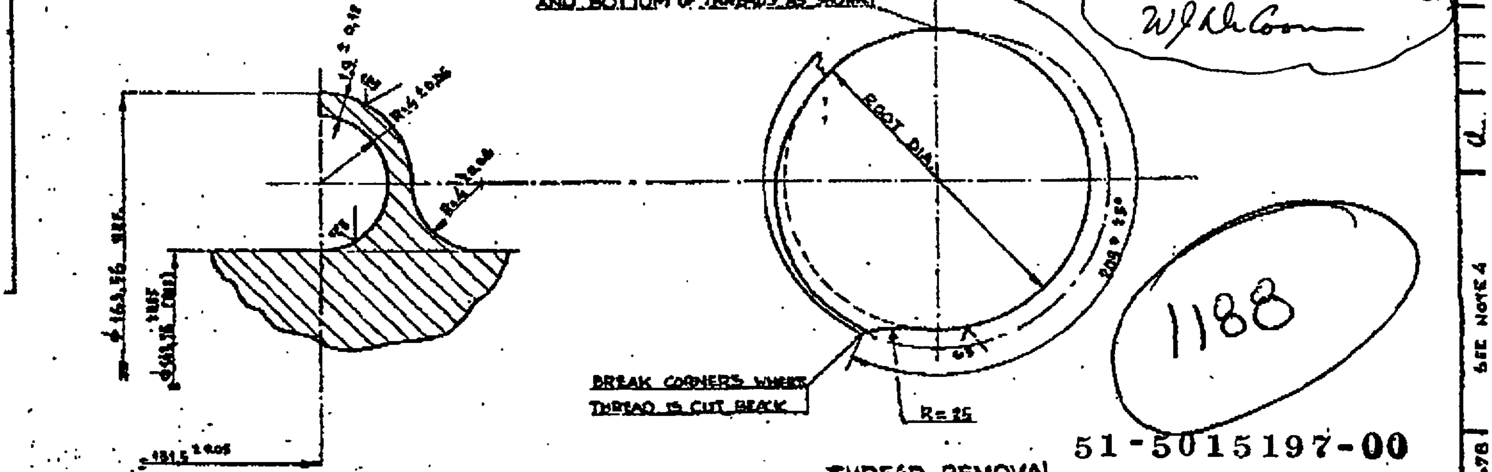
WJ McCoon

BEGINNING AT POINT WHERE THREAD RUNS OUT AT ROOT DIA, REMOVE IMPERFECT THREADS FOR 115° TOP AND BOTTOM OF THREADS AS SHOWN.



THREAD REMOVAL

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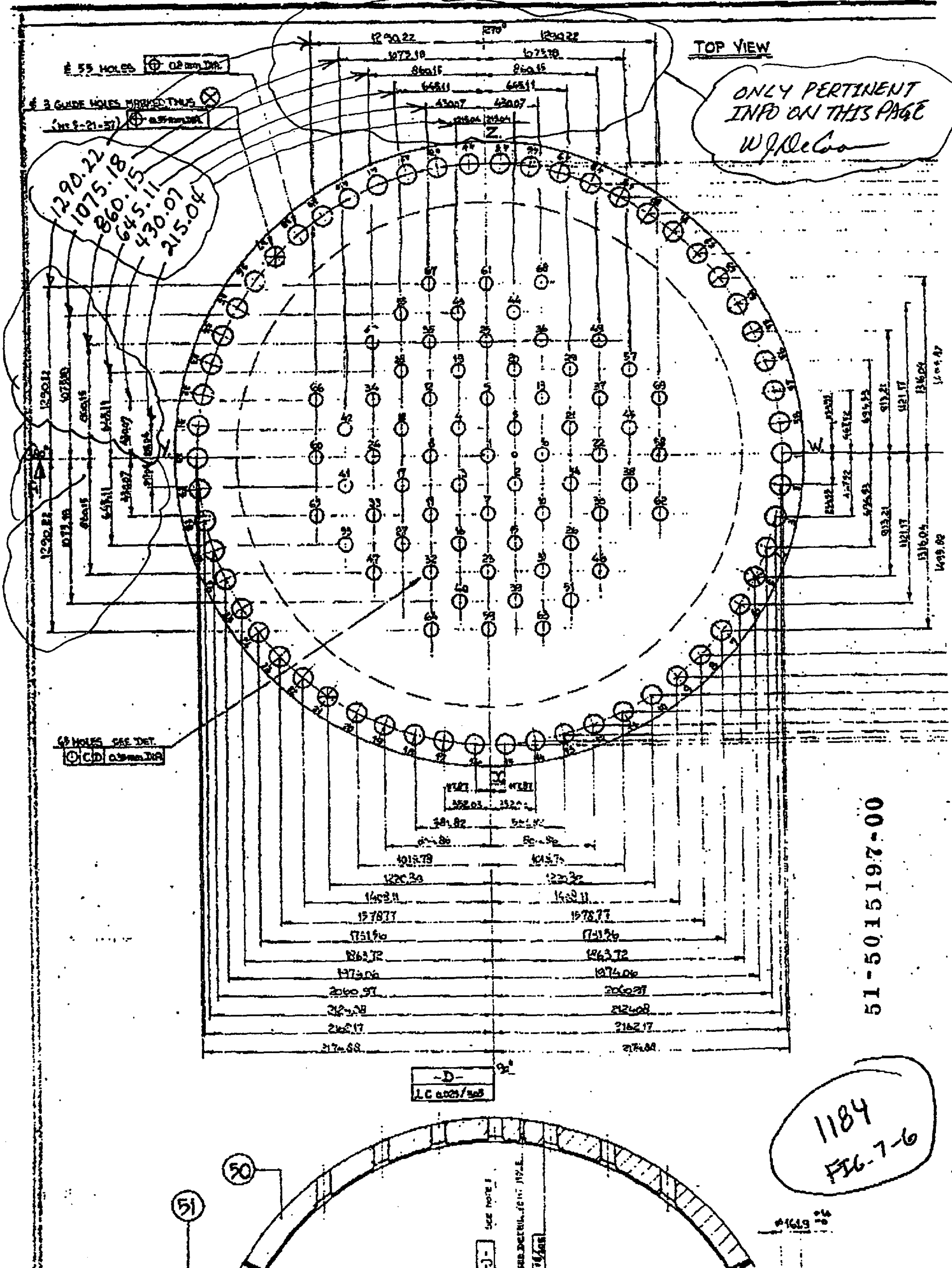
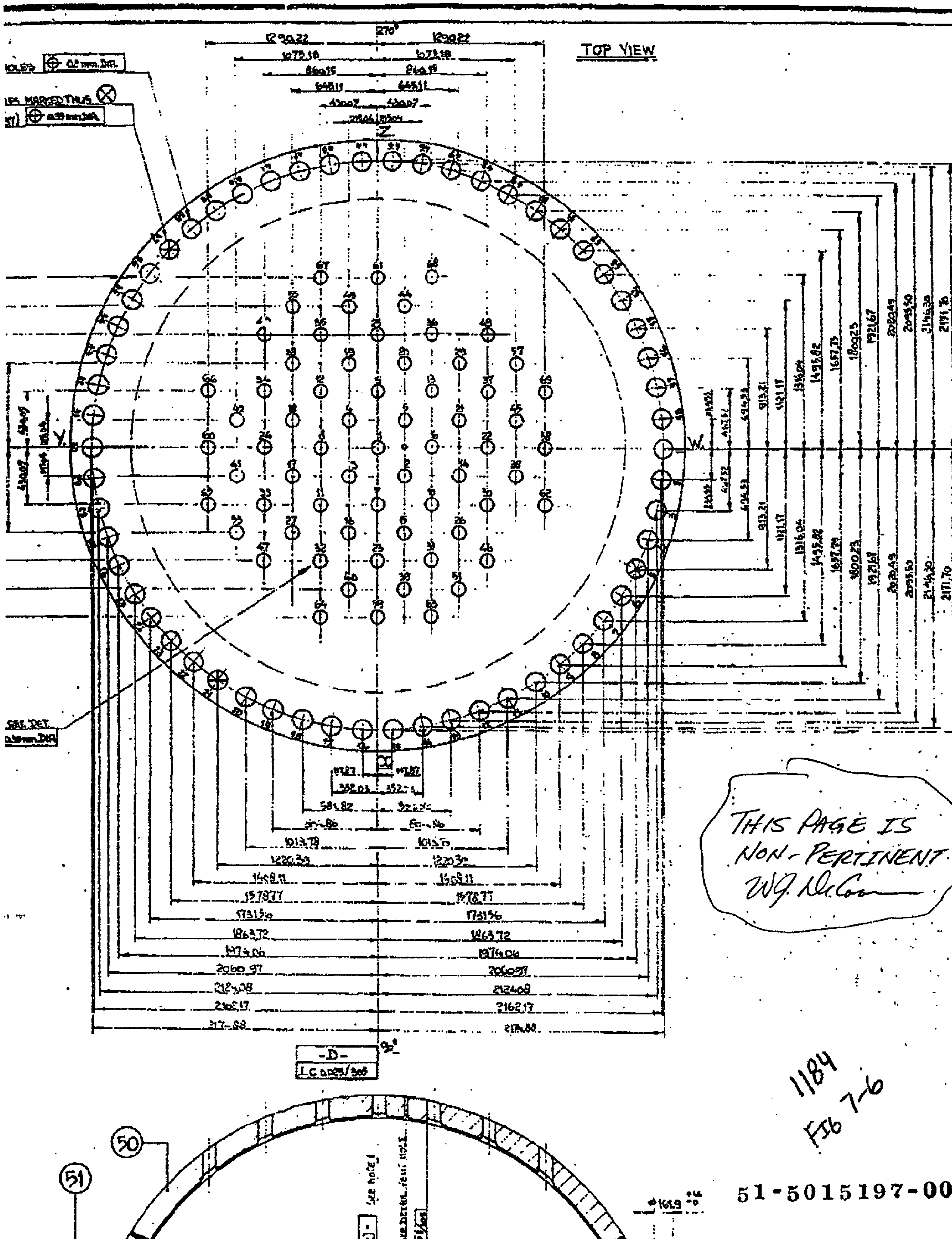
Pg. 9



51-5015197-00 Pg. 10



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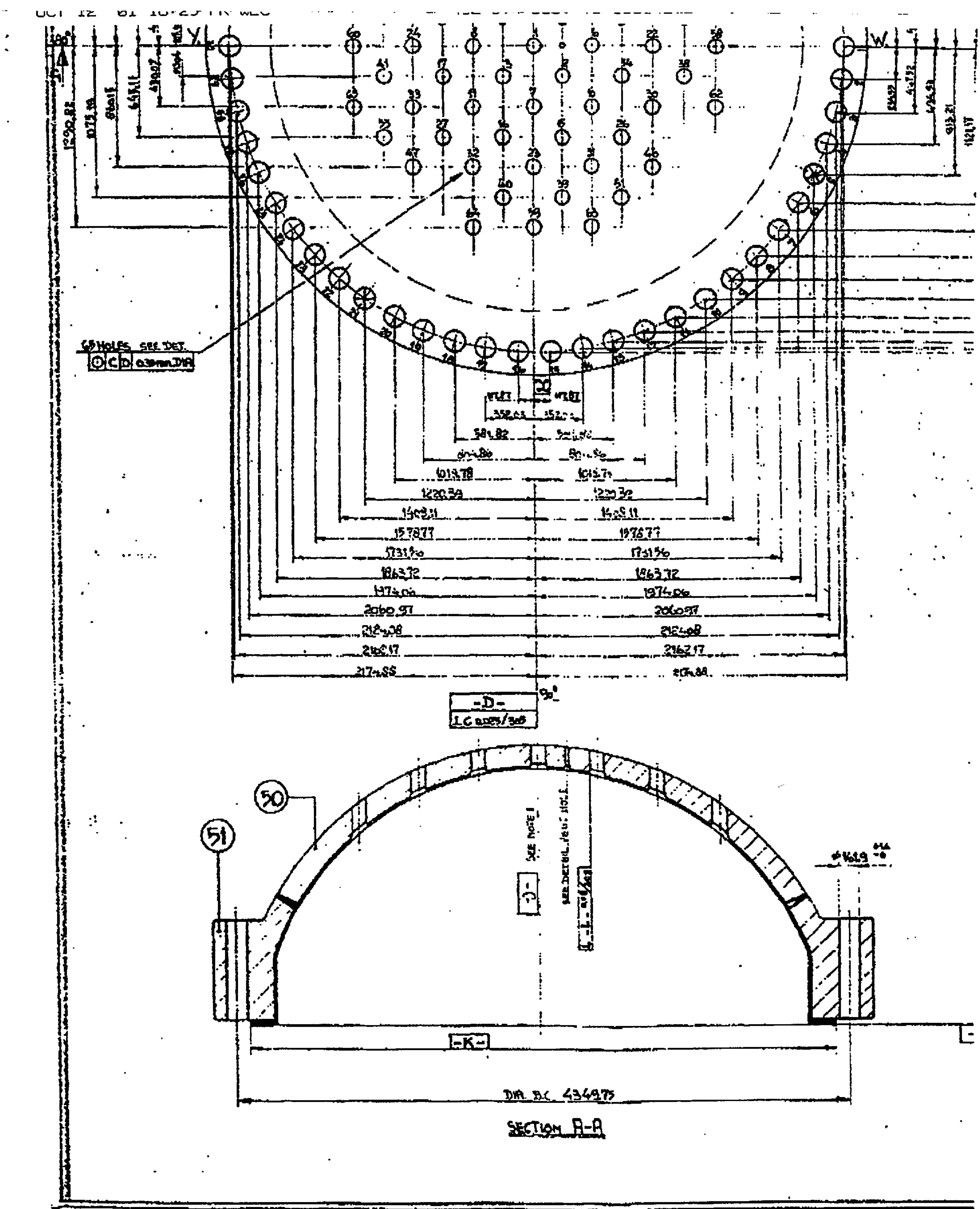
1184
FIG 7-6

51-5015197-00

Pg. 11



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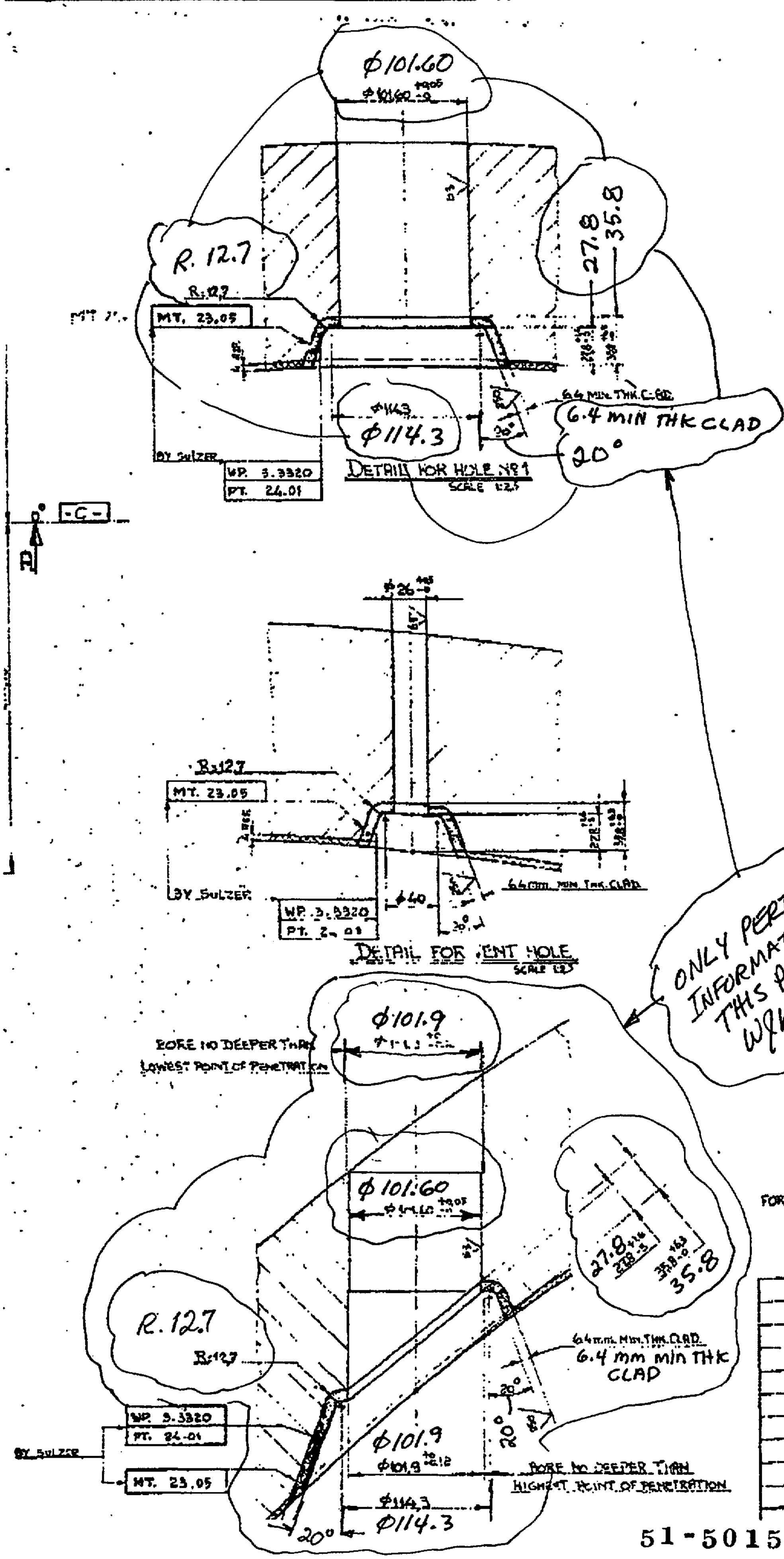
W.G. McG[illegible]

51-5015197-00

1184
7-6

Pg. 12



LETTER	DATUM	OMSCHRIJVING VAN [illegible]
A	9-1-70	CHANGED ACC. PAR. 19

AFGEGEVEN AAN						DATUM
W.E.C.						
LLOYD'S						
APP. BOUW						
SMEDERIJ						
MACH. FABR.						
COÖRD. BUR. W.						
LAB.						
LASSERIJ GTV.						
Mr. HERBERT						
REKENAFD. K.E.						
K.E.						
Mr. RUTHERFORD						
WYZ						

1184

FIG 7-6

NOTES:

1/ CENTERLINE -J- IS DEFINED AS A VERTICAL OF DIR -K- ⊥ L 0.025/305.

2/ ALL INTERNAL MACHINED & CLAD SURFACES UNLESS OTHERWISE NOTED. ALL EXTERNAL SURFACES TO HAVE 500 √RMS OTHERWISE NOTED.

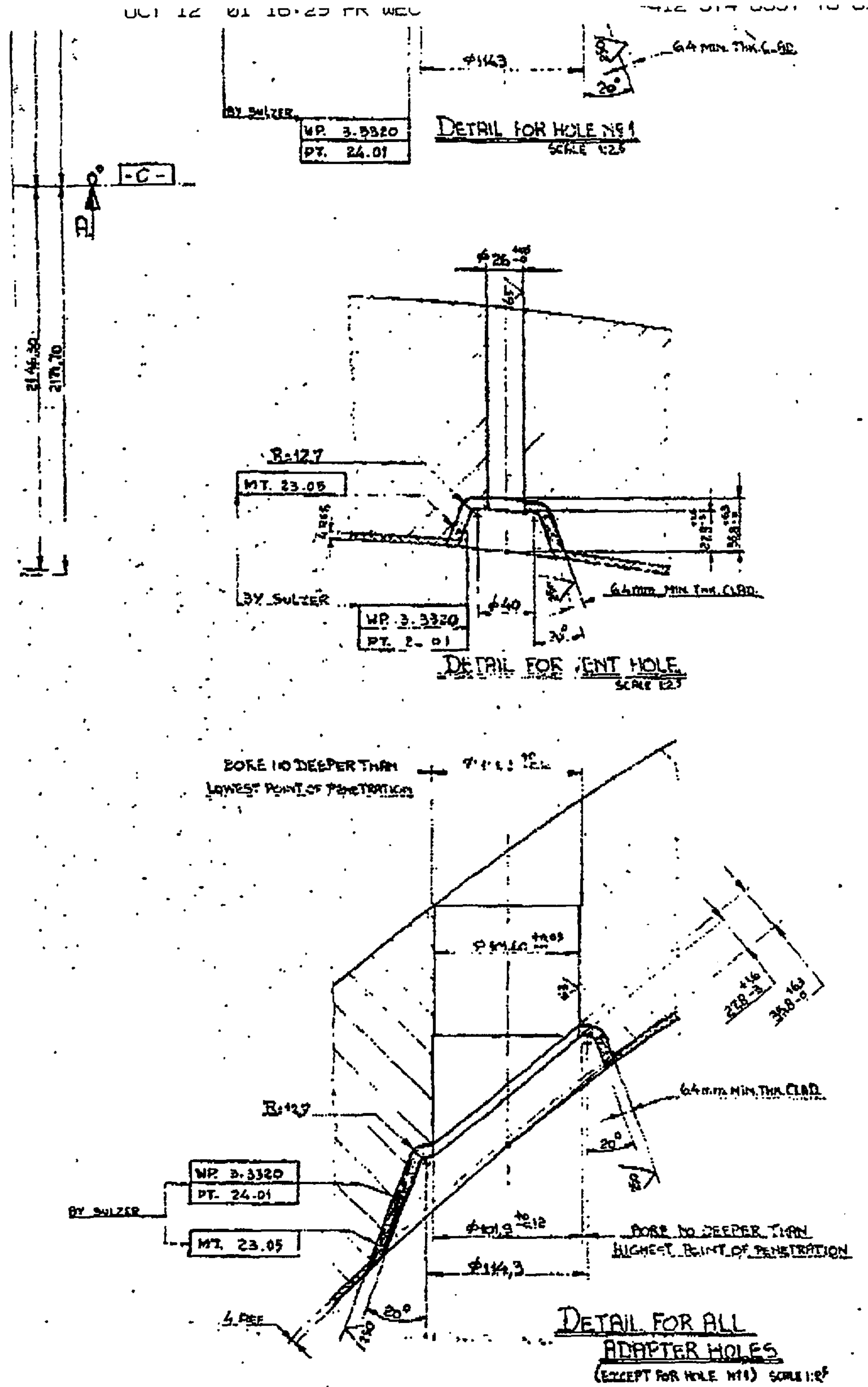
3/ CLOSURE HEAD MANUFACTERED BY SULZER. ACC. TO SULZER D'WG. 0.103.080.218

FOR MACH. DIMENSIONS AND ATTACHMENTS SEE SHEET No.2

VPA-RCPCRV-01

51	1	CLOSURE HEAD FLANGE		R
50	1	CLOSURE HEAD CENTR. DISC.		R
POS.				NORM

51-5015197-00 Pg. 13



NOTES:

1/ CENTERLINE -J- IS DEFINED AS A VERTI... OF DIR -K- ⊥ 0.025/305

2/ ALL INTERNAL MACHINED & CLAD SURFAC... UNLESS OTHERWISE NOTED. ALL EXTERNAL SURFACES TO HAVE ∇ ... OTHERWISE NOTED.

3/ CLOSURE HEAD MANUFACTERED BY SULZE... ACC. TO SULZER DWG. 0.103.080.218

FOR MACH. DIMENSIONS AND ATTACHMENTS SEE SHEET

VPA-RCPLRV-01

MR. HERBERT			
REKENRED K.E			
K.E			
MR. RUTHERFORD			
W.Y.Z.			

POS NR	AANT.	BENAMING	NORM
51	1	CLOSURE HEAD FLANGE	
50	1	CLOSURE HEAD CENTR. DISC.	

STUKLIJST PER — VOOR

DE ROTTERDAMSCHE DR...
AFD. KERNENERG...

157" P.W.R. VESSEL "WE...

CLOSURE HEAD SUB

DRILLING OF ADAPTER HOLE...

GETEKEND	DATUM	SCHAAL
[signature]	19-8-69	1:2.5 1:2.5

SEE NOTE 3 TE VERVAARDIGEN

30678 VOOR O.

TEK. NR. 30678-1184 BLAD 1 VAN 2 A

1184 FIG 7-6

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[signature]

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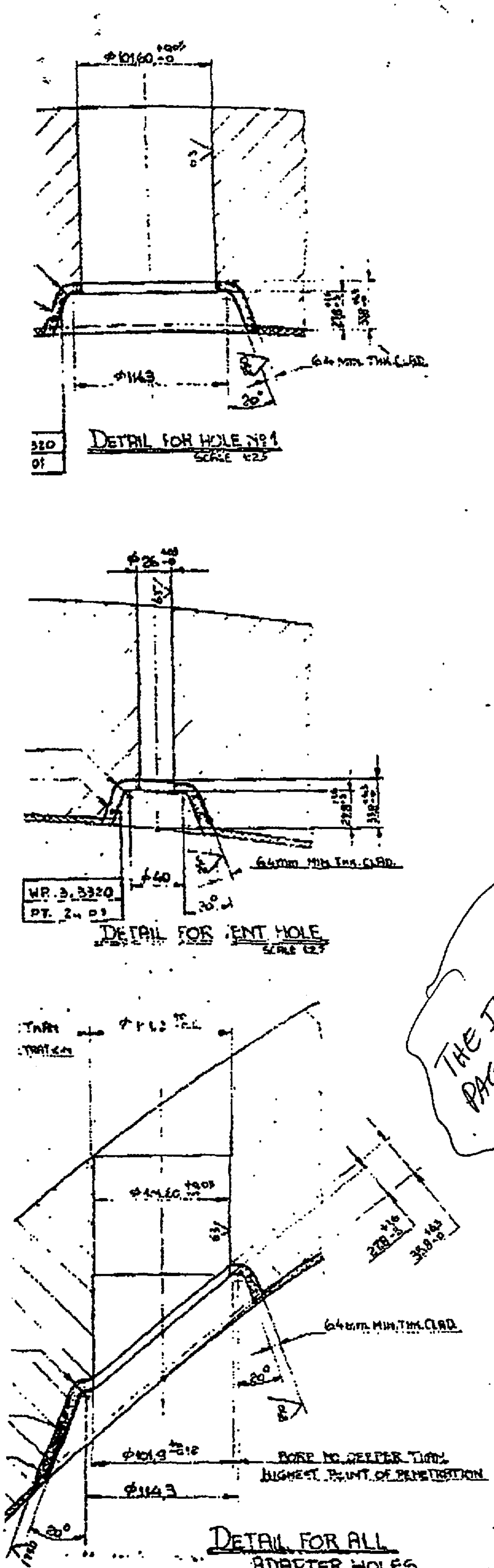

AFGEGEVEN AAN	DATUM										
W.E.C.											
LLOYD'S											
APP. BOUW.											
SMEDERIJ											
MACH. FABR.											
COÖRD. BUR. W.											
LAB.											
LASSERIJ ATV.											
MR. HERBERT											
REKENBED. K.E.											
K.E.											
MR. RUTHERFORD											
WYZ.											

B. DeCooman

NOTES:

1/ CENTERLINE -J- IS DEFINED AS A VERTICAL LINE THRU THE CENTERS OF DIA -K- ⊥ L 0.025/305

2/ ALL INTERNAL MACHINED & CLAD SURFACES TO HAVE 250 √AA FINISH UNLESS OTHERWISE NOTED.
ALL EXTERNAL SURFACES TO HAVE 500 √RMS FINISH UNLESS OTHERWISE NOTED.

3/ CLOSURE HEAD MANUFACTERED BY SULZER.
ACC. TO SULZER DWG. 0.103.080.218

THE INFO ON THIS PAGE IS NON-PERTINENT
WJDG

1184
FIG 7-6

FOR MACH. DIMENSIONS AND ATTACHMENTS SEE SHEET №2

VPA-RCPCRV-01

PAR.	POS NR	AANT.	BENAMING	NORM.	MATERIAAL		BESTELNR.	GEW. IN KG.
	51	1	CLOSURE HEAD FLANGE		ASTM A-508 CL.2			
	50	1	CLOSURE HEAD CENTR. DISC.		ASTM A-533			

STUKLIJST PER | VOOR

J.A. DE ROTTERDAMSCHE DROOGDOK MIJ. N.V.

51-5015197-00

Pg. 15 of 15 Surry

** TOTAL PAGE.15 **

Enclosure 1-3
(Non-Proprietary)

Framatome-ANP Document No. 32-5015624-00, "Surry CRDMH Temperbead Weld Seismic Analysis"

FRAMATOME ANP

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 32 - 5015624 - 00

Title SURRY CRDMH TEMPERBEAD WELD SEISMIC ANALYSIS

PREPARED BY:

NAME D. KIM

SIGNATURE [signature]

TITLE ENGINEER III DATE 11/19/01

COST CENTER 4160048 REF. PAGE(S) 4

REVIEWED BY:

METHOD: ☒ DETAILED CHECK ☐ INDEPENDENT CALCULATION

NAME J. F. SHEPARD

SIGNATURE JF Shepard

TITLE SUPERVISORY ENG DATE 11/19/01

TM STATEMENT: REVIEWER INDEPENDENCE JFS for DKC

PURPOSE AND SUMMARY OF RESULTS:

PURPOSE

The purpose of this document is to check the structural integrity on the Surry CRDMH temperbead weld under seismic condition.

RESULTS

The Surry CRDMH temperbead weld is structurally acceptable under the seismic condition, which is described in Appendix section of Ref. 1.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE/VERSION/REV	CODE/VERSION/REV

THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK

☐ YES ☒ NO

1. PURPOSE

The purpose of this document is to check the structural integrity on the Surry CRDMH temperbead weld under seismic condition.

2. CALCULATION

The following is a calculation of the stresses on the repair weld resulting from OBE and SSE loads. The loads are found at Appendix section of Reference 1. Since a small gap (1 or 2 mils) could exist at operating conditions, no credit is taken for restraint of the Closure head. The bending moments obtained from Reference 1 at the CRDM penetration are:

OBE: M = 29,580 in-lbs
SSE: M = 58,000 in-lbs

The internal pressure is assumed to be equal to 2500 psi.

<u>SSE</u>

Nozzle OD = 4.075 in (Ref. 2)
Nozzle ID = 2.818 in (Ref. 2)

$$t = \frac{1}{2}*(4.075 - 2.818) = 0.6285 \text{ in}$$

$$A = \frac{\pi}{4}*(4.075^2 - 2.818^2) = 6.81 \text{ in}^2$$

$$I = \frac{\pi}{64}*(4.075^4 - 2.818^4) = 10.4 \text{ in}^4$$

$$\sigma_{Bend} = \frac{MR_o}{I} = \frac{58000*2.038}{10.4} = 11.4 \text{ ksi}$$

Pressure Stresses in nozzle:

$$\sigma^P_{Axial} = \frac{PR_i}{2t} = \frac{2500*1.409}{2*0.6285} = 2.8 \text{ ksi}$$

$$\sigma^P_{Hoop} = 2*2.8 = 5.6 \text{ ksi}$$

$$\sigma^P_{Radial} = -P/2 = -1.25 \text{ ksi}$$

$\sigma_L = \sigma_{Bend} + \sigma^P_{Axial} = 11.4 + 2.8 = 14.2$ ksi

$\sigma_{Hoop} = 5.6$ ksi

$\sigma_{Radial} = -1.25$ ksi

Stress Intensity = 14.2 – (-1.25) = 15.45 ksi

Allowable Stress Intensity (Section III, Appendix of Ref. 3)

= Lesser of 2.4 S_m or 0.7 S_u
= 2.4*23.3 = 55.9 ksi or =0.7*80 = 56.0 ksi
= 55.9 ksi

Therefore, comparing SI and the allowable, the SSE load is acceptable.

<u>OBE</u>

The bending stress is 0.51*SSE stress

$\sigma_{Bend} = 0.51*11.4 = 5.81$ ksi

$\sigma_L = \sigma_{Bend} + \sigma^P_{Axial} = 5.81 + 2.8 = 8.6$ ksi

$\sigma_{Hoop} = 5.6$ ksi

$\sigma_{Radial} = -1.25$ ksi

Stress Intensity = 8.6 – (-1.25) = 9.85 ksi

Allowable Stress Intensity = 1.5 S_m = 1.5*23.3 = 35 ksi (assume Level B)

Thus, the OBE load is acceptable.

3. CONCLUSION

The Surry CRDMH temperbead weld is structurally acceptable under the seismic condition, which is described in Appendix section of Ref. 1.

4. REFERENCES

1) FRA-ANP Doc. 51-5015050-02, "Surry CRDM Nozzle ID Temper Bead Weld Repair Requirements"

2) FRA-ANP Dwg. 02-5015149E-02, "Surry 1&2 CRDM Nozzle ID Temper Bead Weld Repair"

3) 1989 ASME BOILER AND PRESSURE VESSEL CODE with no addenda

Attachment 4

(Non-Proprietary - Redacted)

Summary of Structural Evaluation of Weld Repair of CRDM Housings
(Non-Proprietary)
with the following enclosures:

Turkey Point CRDM Temperbead Bore Weld Analysis (Redacted)
Surry 1 and 2 Reconciliation with Turkey Point 3 RV Head and CRM Nozzles (Non-Proprietary)
Surry CRDMH Temperbead Weld Seismic Analysis (Non-Proprietary)
Surry CRDM Nozzle IDTB weld Anomaly flaw Evaluation (Redacted)
Surry CRDM Nozzle 1.0" J-Groove Weld Flaw Evaluation (Redacted)
Surry CRDM J-Groove Weld Stress For Flaw Growth (1" Chamfer) (Redacted)

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

SUMMARY OF STRUCTURAL EVALUATION OF WELD REPAIR OF CRDM HOUSINGS SURRY POWER STATION UNIT 1

0.1 OBJECTIVE:

The objective of this summary is to document the review of the structural evaluation of the repair of the following six CRDM housings on the reactor head of Surry Power Station Unit 1: S-1-18, S-1-27, S-1-40, S-1-47, S-1-65, S-1-69.

0.2 INTRODUCTION AND BACKGROUND:

Due to the recent experience of degradation of CRDM nozzle housing in the vicinity of the J-groove weld to the reactor vessel head described in NRC Bulletin 2001-01, Dominion has inspected the CRDM housing penetrations to the reactor head for Surry Unit 1. The inspection revealed evidence of degradation at the J-Groove weld and possible leakage at the six CRDM housing penetrations cited above. Framatome ANP was contracted by Dominion to repair the nozzles.

Repair has been performed to meet the applicable configuration requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NB, 1989 edition. The repair weld has been deposited using the machine GTAW process with cold wire feed, in accordance with the ASME Section XI, IWA-4000 with modification as described in by Relief Requests SR-27 and SR-28.

The repair effort followed several steps, not necessarily in the order given below. A baseline volumetric and surface examination was performed for the repair region. The lower portion of the thermal sleeve was cut and removed with automatic tools after cleaning. The CRDM nozzle was rolled into the reactor vessel head penetration. The lower end of the nozzle was machined away into the head to make the weld preparation beyond the degraded area. The J-weld at the bottom end of the penetration was chamfered by grinding to remove part of the degraded weld. The bored region of the head and weld prep on the bottom of the remaining portion of the CRDM nozzle were examined by PT. The repair area was cleaned for welding and weld material was deposited. The repair weld was machined to reestablish a nozzle free path and to provide a suitable surface for PT and UT. PT and UT examinations were performed for the repair. The repair was remediated using an abrasive water-jet. The thermal sleeve was replaced as the last step of the repair.

The portion of the reactor vessel head (RVH) containing the CRDM nozzle is fabricated from SA-533 Grade B, Class 1. The portion of the CRDM nozzle that penetrates the RVH is SB-167 Alloy 600. The weld material for the repair is ERNiCrFe-7, UNS N06052. The cobalt content of the weld filler material was limited to 0.2%. The replacement thermal sleeve has been welded to the upper sleeve using metal insert in accordance with SFA 5.9 ER309L or ER316L per ASME Section II.

Three different structural evaluations have been performed to establish the structural integrity of the repair and design life of the repair:

1) Stress analysis of the repair has been performed conforming to the requirements

of ASME, Section III, Subsection NB, Paragraph NB-3000, 1989 Edition.

2) A fracture mechanics analysis has been performed in accordance with IWB-3132.4 and IWB-3600 of ASME Section XI Code. This analysis considered a 0.100-inch weld anomaly and assumed it to be a linear defect and extending into the repair weld in any direction at the triple point. The triple point is defined as the intersection of the reactor head base material, the CRDM nozzle, and the repair weld. It has been justified by experience that the assumed flaw is bounding.

3) A fracture mechanics analysis has also been performed to justify a postulated flaw remaining in the J-groove weld remnant between the original CRDM nozzle and the reactor vessel head. This analysis is important because the flaw in the remaining weld cannot be characterized by available NDE methods. The size of the flaw considered in this analysis is equal to the largest radial length through the remaining J-weld. The flaw growth analysis has been used as one of the considerations to establish design life of the repair.

These three analyses are summarized below. The summary includes the configurations analyzed, loading conditions, design criteria, and code compliance. The details of stresses, cumulative usage factors, flaw tolerance and flaw growth analyses are presented. Based upon the results of these conservative analyses, the design life of the repair is predicted to be at least five years. The life of the repair is dependent on the size of the remaining J-groove weld, where the analysis conservatively postulated an initial flaw through the remaining thickness of the weld.

1. ASME SECTION III ANALYSIS OF REPAIR

1.0 OBJECTIVE

The purpose of this review is to summarize the ASME Section III analyses that have been performed for the CRDM temperbead bore weld repair for Surry Unit 1 Reactor Vessel Upper Head Penetrations S-1-18, -27, -40, -47, -65, and -69. The repair consists of cutting the CRDM housing above the original attachment weld, removing the lower portion of the housing and welding the remaining housing to the reactor vessel upper head with a temperbead weld. Analyses have been performed that demonstrate that the repair design meets the applicable requirements of the ASME Code Section III. The Surry CRDM nozzles are similar to corresponding nozzles analyzed previously for this repair procedure. A formal reconciliation was performed to allow use of these previous analyses for Surry.

1.1 GEOMETRY/FINITE ELEMENT MODEL DEFINITION

The finite element model used to analyze the CRDM housing nozzle to reactor vessel upper head weld region is documented in References 1-1, 1-2, and 1-3. The finite element model is a 3-dimensional model of a 180-degree segment of a CRDM tube with the adjacent head region and interconnecting weld.

1.2 MATERIALS

The materials of the components in the finite element model are summarized below (References 1-1, 1-2, and 1-3):

Reactor Vessel Head Base Metal = ASTM A533, Grade B, Class 1 (Mn-Mo Steel)
CRDM Housing Nozzle = ASME SB-167 Inconel
Cladding = Stainless Steel
J-Groove Buttering = Alloy 600 (Inconel)
J-Groove Filler = Alloy 600 (Inconel)
Repair Weld = ERNiCrFe-7, UNS N06052 Per ASME Section II, Part C, SFA-5.14, with properties similar to Alloy 690.

1.3 LOADS

The loads considered in the design of the CRDM IDTB (ID Temperbead) weld repair are based on those considered in the original design specification (Reference 1-5) and design report (Reference 1-6) for the reactor vessel top head and CRDM housings. The loads considered are:

Design Pressure/Temperature
Plant heatup and cooldown at 100ºF/hr.
Plant loading and unloading at 5% of full power per minute
Small step load increase and decrease
Large step load decrease
Loss of load
Loss of power
Loss of Flow
Reactor Trip from full power
Turbine roll test
Primary side hydrostatic test at 3105 psig
Primary side hydrostatic test at 2485 psig
Steady state fluctuations
Steam pipe break (faulted)
OBE seismic loading
DBE seismic loading

For analysis purposes, operational transients have been grouped into three separate analyses: 1) heatup/cooldown, 2) plant loading/unloading, and 3) remaining (or rapid) transients. For the plant loading/unloading transient, the ASME Section III fatigue evaluation for the IDTB weld repair has assumed a total of 14,500 loading/unloading events over the plant design life. While this assumption does not bound the 29,000 cycles assumed in the original design specification, it is bounding relative to actual plant operation. The 29,000 cycles of loading and unloading was based on load-following operation. Surry has operated (and will continue to operate) in a base-load capacity manner, which results in significantly fewer loading/unloading cycles. The assumed value of 14,500 cycles is still very conservative. The rapid transient has been defined to bound the small step increase/decrease, large step load decrease, loss of load, loss of power, loss of flow, and reactor trip operational transients. The transients used in the analyses have been reviewed and determined to envelop the design transients for Surry.

1.4 LOADING CONDITIONS/ STRESS CRITERIA:

The following loading conditions and stress criteria are used in the evaluation documented in Reference 1-3. The 1989 Edition of the ASME Code (No Addenda), Section III (Reference 1-4) is used for the evaluation.

Primary Stress Intensities for Design Conditions:

NB-3221.1, Primary General Membrane Stress Intensity ($P_m \leq S_m$)

NB-3221.2, Local Membrane Stress Intensity ($P_l \leq 1.5\ S_m$)

NB-3221.3, Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 1.5\ S_m$)

Primary + Secondary Stress Intensity Range for Service Level A/B (normal/upset) Conditions:

NB-3222.2, Primary + Secondary Stress Intensity Range (P + S Stress Intensity Range $\leq 3\ S_m$)

Fatigue Usage

NB-3222.4, Fatigue Usage $\leq$ 1.0

Primary Stress Intensities for Emergency (Level C) Conditions:

NB-3224.1, Primary General Membrane Stress Intensity ($P_m \leq 1.2\ S_m$)

NB-3224.1, Local Membrane Stress Intensity ($P_l \leq 1.8\ S_m$)

NB-3224.1, Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 1.8\ S_m$)

Primary Stress Intensities for Faulted (Level D) Conditions:

NB-3225, F-1331.1(a), Primary General Membrane Stress Intensity ($P_m \leq 0.7\ S_u$)

NB-3225, F-1331.1(b), Local Membrane Stress Intensity ($P_l \leq 1.05\ S_u$)

NB-3225, F-1331.1(c), Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 1.05\ S_u$)

Primary Stress Intensities for Test Conditions:

NB-3226(a), Primary General Membrane Stress Intensity ($P_m \leq 0.9\ S_y$)

NB-3226(b), Primary Membrane + Primary Bending Stress Intensity ($P_l + P_b \leq 2.15\ S_y - 1.2P_m$)

The repair is analyzed to 1989 version of ASME Section III Code (Reference 1-4). The original stress report (Reference 1-6) was prepared conforming to the requirements of 1968 version of ASME Section III Code (Reference 1-8). The stress criteria of the original design differ from the 1989 version of Section III Code only for allowable stresses in OBE and SSE conditions. In the original design, the stress in the OBE condition was checked against an allowable stress intensity of 1.2 S_m and SSE condition was checked against an allowable stress intensity of 1.8 S_m. In order to comply with the original design criteria, the stresses under seismic loading (Reference 1-7) were also compared with the original Code allowable.

1.5 RESULTS:

The results of the ASME Section III analysis of the weld repair are summarized below:

Primary Stress Intensities for Design Conditions (Design Pressure at Design Temperature):

RV Head: $P_m = 16.6$ ksi $\leq S_m = 26.7$ ksi
$P_l = 20.4$ ksi $\leq 1.5\ S_m = 40.1$ ksi
$P_l + P_b = 25.6$ ksi $\leq 1.5\ S_m = 40.1$ ksi

Nozzle/Weld: $P_m = 6.2$ ksi $\leq S_m = 23.3$ ksi
$P_l = 9.85 \leq 1.5\ S_m = 35.0$ ksi
$P_l + P_b = 9.85$ ksi $\leq 1.5\ S_m = 35.0$ ksi
(Also less than 1.2 S_m=27.96 ksi)

Normal/Upset Service Level (A/B) Condition

Primary + Secondary Stress Intensity Range:
Heatup/Cooldown Transient: $S_n = 36.7$ ksi $\leq 3\ S_m = 80.0$ ksi
Loading/Unloading Transient: $S_n = 16.1$ ksi $\leq 3\ S_m = 80.0$ ksi
Rapid (Remaining) Transient: $S_n = 9.1$ ksi $\leq 3\ S_m = 80.0$ ksi

Fatigue Usage

The total fatigue usage, based on an assumed fatigue strength reduction factor of 4.0, for a 14-year service life is calculated to be 0.525. With this result, the qualified operating life for which the fatigue usage is less than 1.0 is 26.7 years.

Emergency (Level C) Conditions:

RV Head:
Maximum Allowable Pressure Based on P_m Limit = 4,819 psi
Maximum Allowable Pressure Based on P_l Limit = 5,895 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 4,697 psi

Nozzle/Weld:
Maximum Allowable Pressure Based on P_m Limit = 11,089 psi
Maximum Allowable Pressure Based on P_l Limit = 16,633 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 16,633 psi

All of the maximum allowable pressures based on the Emergency (Level C) condition stress limits are greater than the maximum hydrotest pressure of 3105 psi. The level C pressure loading is not specified for Surry.

Faulted (Level D) Conditions:

RV Head:
Maximum Allowable Pressure Based on P_m Limit = 8434 psi
Maximum Allowable Pressure Based on P_l Limit = 10,294 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 8,203 psi

Nozzle/Weld:
Maximum Allowable Pressure Based on P_m Limit = 22,540 psi
Maximum Allowable Pressure Based on P_l Limit = 33,830 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 33,830 psi

All of the maximum allowable pressures based on the Faulted (Level D) condition stress limits are greater than the maximum hydrotest pressure of 3,105 psi. The level D pressure loading is not specified for Surry.

Primary Stress Intensities in SSE Condition

RV Head:
Insignificant Seismic effect

Nozzle/Weld:
$P_l + P_b$ = 15.45 ksi $\leq 2.4\ S_m$= 55.9 ksi.
(Also $\leq 1.8\ S_m$= 41.94 ksi)

Test Conditions:

RV Head:
Maximum Allowable Pressure Based on P_m Limit = 6,777 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 5,225 psi

Nozzle/Weld:

Maximum Allowable Pressure Based on P_m Limit = 12,702 psi
Maximum Allowable Pressure Based on $P_l + P_b$ Limit = 15,121 psi

All of the maximum allowable pressures based on the test condition stress limits are greater than the maximum hydrotest pressure of 3,105 psi. No hydrotest of this level is planned for Surry.

1.6 CONCLUSION:

The CRDM housing nozzle temperbead weld repair design meets the stress and fatigue requirements of the ASME Code, Section III, 1989 edition w/o Addenda. The conservative fatigue analysis indicates that the repair design has a qualified operating life of at least 26.7 years.

1.7 REFERENCES:

1-1 Framatome ANP Document No. 32-5014129-00, "Turkey Point - CRDMH 3D FE Model."
1-2 Framatome ANP Document No. 51-5015197-01, "Surry 1 & 2 Reconciliation with Turkey Point 3 RV HD & CRM Noz." (Included as Enclosure 1-2)
1-3 Framatome ANP Document No. 32-5014640-00, "Turkey Point – CRDM Temperbead Bore Weld Analysis." (Included as Enclosure 1-1)
1-4 ASME Boiler and Pressure Vessel Code, 1989 Edition, Section III, No Addenda
1-5 Surry Reactor Vessel Design Specification 676499, Rev. 1, "Addendum to Equipment Specification 676413, Rev. 1, Surry Power Station 1."
1-6 Calculation 30678-1130, "Reactor Vessel – Final Stress Report (Parts I & II), Surry Power Station Units 1 and 2, " Rotterdam Dockyard Company.
1-7 Framatome ANP Document No. 32-5015624-00, "Surry CRDMH Temperbead Weld Seismic Analysis." (Included as Enclosure 1-3)
1-8 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1968 Edition to and including Winter 1968 Addenda

2. SURRY CRDM NOZZLE IDTB WELD ANOMALY FLAW EVALUATIONS

2.1 PURPOSE:

This review summarizes the CRDM nozzle IDTB weld anomaly flaw evaluation. This is a common evaluation for IDTB weld repair performed on the following six CRDM nozzles of Surry Power Station Unit-1: S-1-18, S-1-27, S-1-40, S-1-47, S-1-65, and S-1-69.

2.2 CONFIGURATION:

A fracture mechanics evaluation has been performed for a postulated weld anomaly in the CRDM nozzle IDTB weld repair design (Reference 2-1). During the welding process a maximum of 0.1" weld anomaly may be formed due to lack of fusion at the triple point.

The postulated weld anomaly is a 0.1" semi-circular region of lack of fusion extending 360-degrees around the circumference at the triple point location at the intersection of three materials: the Alloy 600 nozzle, the Alloy 52 weld, and alloy steel head. The flaw evaluation simulated the defect as a 360-degree circumferential crack of depth of 0.1" on the OD of a circular tube. The evaluation also postulated an axially oriented semi-circular OD surface flaw with depth equal to 0.1" and axial length of the flaw equal to 0.2". Both of these circumferential and axial flaws postulated on the outer surface propagate horizontally into the weld material. A semi-circular, cylindrically oriented flaw is also postulated along the interface between the weld and head, and propagates downward between the two components. The finished thickness of the wall used in the analysis is 0.488".

2.3 MATERIAL PROPERTIES:

Fracture toughness curves for SA-533 Grade B, Class 1 material are illustrated in the ASME Section XI, Code, 1989 in Figure A-4200-1. At an operating temperature of 600°F, the K_{Ia} fracture toughness value for this material is above 200 ksi$\sqrt{in}$ for assumed RT_{NDT} of 60°F. The toughness properties of Alloy 600 and weld material are better than 200 ksi$\sqrt{in}$ and; therefore, an upper-shelf value of 200 ksi$\sqrt{in}$ is used in the analysis (Reference 2-1).

2.4 LOADS:

The transient loads applicable for evaluation of this repair were conservatively grouped into three categories:

Heatup/Cooldown	3.33 cycles per year
Plant Loading/Unloading	250 cycles/year
Remaining rapid transients	46.67 cycles per year

2.5 APPLICABLE CRITERIA:

The flaw acceptance is based on the 1989 ASME Code Section XI criteria for applied stress intensity factor (IWB-3612) and limit load (IWB-3642). For flaw growth analysis in the RV Head, Article A-4300 of Section XI code is used. For flaw growth rate in the repair weld Article C-3210 of Section XI (normally applicable to austenitic stainless steel in an air environment) has been used.

2.6 RESULTS:

The results of the analyses showed:
A minimum fracture toughness margin of 11.4 compared to the required margin of $\sqrt{10}$ per IWB-36-12.

A margin on limit load of 6.25, compared to the required margin of 3.0 per IWB-3642.

Fatigue crack growth is minimal. The predicted crack growth over 25 years is from 0.100" to 0.114". There is no acceptance standard for this. However, the predicted crack will still remain shallow. (Details of evaluation are provided in Enclosure 2-1.)

2.7 CONCLUSION:

The IDTB weld repair will maintain structural integrity for the predicted life of repair.

2.8 REFERENCE:

2-1 Framatome ANP, Document No. 32-5015219-00, "SURRY CRDM NOZZLE IDTB WELD ANOMALY FLAW EVALUATIONS." (Included as Enclosure 2-1)

3. FLAW EVALUATION OF THE REMAINING J-GROOVE WELD

3.1 OBJECTIVE:

The purpose of this review is to summarize the flaw evaluation of the remaining J-groove weld following the IDTB weld repair of the following six CRDM nozzles of Surry Power Station Unit-1: S-1-18, S-1-27, S-1-40, S-1-47, S-1-65, and S-1-69.

3.2 BACKGROUND:

Since a potential flaw in the J-groove weld cannot be sized by currently available NDE techniques, it must be assumed that the as-left condition of the remaining J-groove weld includes degraded or cracked weld material extending through the entire J-groove weld and Alloy 182 butter material.

The hoop stresses in the J-groove weld are generally about twice the axial stress; therefore, the preferential direction for cracking is radial out from the bore radius. It is postulated that a radial crack in the Alloy 182 weld metal would propagate through the weld and butter, to the interface with the low-alloy steel head. Extensive industry experience has shown that flaws originating in an Alloy 82/182 weld have not propagated into the ferritic base material, and it is fully expected that such a crack would then blunt and arrest at the butter-to-head interface. However, for this evaluation, it is conservatively assumed that the stress corrosion crack in the weld would combine with a small flaw in the reactor head steel to form a large radial corner flaw that would propagate into the low alloy head by fatigue crack growth under cyclic loading conditions.

3.3 CONFIGURATION:

Analytically, this flaw has been simulated using a corner flaw model (Reference 3-1). The repair incorporates a chamfer at the inside corner of the remnant J-groove weld to limit the potential crack length through the weld from the inside corner of the bore chamfer to the low alloy steel vessel head. The evaluation assumes the initial flaw depth as 1.053 inch, which represents the distance completely through the remaining weld.

3.4 MATERIAL PROPERTIES:

Fracture toughness curves for SA-533 Grade B, Class 1 material are illustrated in the ASME Section XI, Code, 1989 in Figure A-4200-1. At an operating temperature of 600°F, the K_{Ia} fracture toughness value for this material is above 200 ksi√in for assumed RT_{NDT} of 60°F. The toughness properties of Alloy 600 and weld material are better than 200 ksi√in and; therefore, an upper-shelf value of 200 ksi√in is used in the analysis.

3.5 APPLICABLE CRITERIA:

The flaw acceptance is based on the 1989 ASME Code Section XI criteria for applied stress intensity factor (IWB-3612).

3.6 LOADINGS:

The imposed stress distribution was obtained from a 3-D ANSYS finite element analysis, which was performed to determine operating transient stresses in the vicinity of the CRDM nozzle following the repair (Reference 3-2). Previous analyses had determined that the outermost nozzles with the largest "hillside angle" (the relative angle between the local plane of the reactor head and the nozzle vertical centerline) experience the greatest increase in stress in the region of the J-groove weld. Therefore, the finite element model represented one of the outermost nozzles, and the results will conservatively bound all nozzle locations that have a smaller hillside angle. The finite element analysis found that the highest stresses occur at the uphill side of the nozzle along the vertical plane formed by the centerlines of the nozzle and the reactor. Transient analyses were performed for normal heatup and cooldown cycles, plant loading and unloading cycles, reactor trip, and other rapid transients. The maximum stresses were determined along a line into the reactor head material from the uphill "corner" of the nozzle bore, representing the progression of the crack front of the assumed corner crack.

Residual stresses were not explicitly included in this flaw evaluation, since a crack that has propagated all the way through the weld would tend to relieve these stresses, and a crack at the butter-to-head interface would experience only compressive residual stress ahead of the crack.

The fracture mechanics analysis was performed assuming the following pattern for accumulating cycles:

Transient	Frequency (cycles / year)
Heat up / Cool down	3.33
Plant Loading / Unloading	50.00
Large Step Decrease	3.33
Loss of Load	1.33
Loss of Flow	1.33
Reactor Trip	6.67
Remaining Transients	34.00

* The original design specification included 29,000 cycles of plant loading/unloading for the life of the plant. As discussed previously, the number of cycles in the design specification was conservatively based on load-following operation. The 50 cycles/year is conservative for the actual base load capacity mode of operation under which Surry has operated and will continue to operate.

3.7 RESULTS:

The crack growth analysis was performed for each set of transients for each year and iteratively summed by linking the incremental crack growth for each of the sets of transients for each year. The results are compared to the fracture toughness requirements of Section XI. Applying the conservatively assumed number of cycles per year, the fracture mechanics analysis shows that the crack will be acceptable for over five years of operation. The flaw depth at the end of five years is projected to be 1.123". The calculated stress intensity factor at the final flaw size for the most severe transient is less than K_I = 63.16 ksi -√ in, compared to the fracture toughness upper-shelf value of K_{Ia} = 200.0 ksi -√ in. This provides a safety margin of 3.17, which is greater than √10 safety margin required by Article IWB-3612 of the Code.

(Details of the fracture mechanics analysis are given in Enclosure 3-1. Information on the stress analysis is provided in Enclosure 3-2.)

3.8 REFERENCES:

3-1 Framatome ANP Document No. 32-5015650-00, "SURRY CRDM NOZZLE 1.0" J-GROOVE WELD FLAW EVALUATION." (Included as Enclosure 3-1)
3-2 Framatome ANP Document No. 32-5015651-00, " SURRY-CRDMH J-GROOVE WELD STRESS FOR FLAW GROWTH," (1" CHAMFER), (Included as Enclosure 3-2)

Enclosure 1-1
(Redacted)

Framatome ANP Document No. 32-5014640-01,
"Turkey Point – CRDM Temperbead Bore Weld Analysis."

FRAMATOME ANP

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 32 - 5014640 - 01

Title TURKEY POINT - CRDM TEMPERBEAD BORE WELD ANALYSIS

PREPARED BY:		REVIEWED BY:	
		METHOD: ☒ DETAILED CHECK ☐ INDEPENDENT CALCULATION	
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TITLE ENG III/ENG III	DATE 11/28/01	TITLE ADVSRY ENG	DATE 11-29-01
COST CENTER 41020	REF. PAGE(S) 30	TM STATEMENT: REVIEWER INDEPENDENCE	HDM

PURPOSE AND SUMMARY OF RESULTS:

THIS IS THE NON-PROPRIETARY VERSION OF 32-5014640-00.

Purpose:
The purpose of this calculation is to analyze the CRDMH nozzle temperbead weld repair design described in Reference 2. This repair consists of cutting the CRDM housing above the original attachment weld, removing the lower portion of the housing and welding the remaining housing to the RV head with a temperbead weld.

Conclusion :
The calculations herein demonstrate that the CRDMH Nozzle temperbead repair design meets the stress and fatigue requirements of the Design Code (ASME Code, Section III, 1989 edition w/o addendum – Reference 4). The conservative fatigue analysis indicates that the repair design is acceptable for at least [ ] years of operation.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:		THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK
CODE/VERSION/REV	CODE/VERSION/REV	☐ YES ☒ NO

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RECORD OF REVISIONS		
REVISION	DESCRIPTION	DATE
00	ORIGINAL RELEASE	10/01
01	NON-PROPRIETARY VERSION OF ORIGINAL DOC	11/01

1.0 Purpose

The purpose of this calculation is to analyze the CRDMH nozzle temperbead weld repair design described in Reference 2. This repair consists of cutting the CRDM housing above the original attachment weld, removing the lower portion of the housing and welding the remaining housing to the RV head with a temperbead weld.

This calculation will demonstrate that the design meets the applicable requirements of the ASME Code, Section III. Installation of this repair may result in a given closure head assembly having CRDMs with both the repair design and the original design. Therefore, this document (an analysis of the repair design) is considered as a supplemental analysis to the original stress report (an analysis of the original design).

2.0 Background

In December 2000, inspection of the Alloy 600 control rod drive mechanism (CRDM) nozzle penetrations in the RV closure head (RVH) at Oconee Unit 1 identified leakage in the region of the partial penetration attachment weld between the RVH and the CRDM nozzle. This leakage, identified as the result of Primary Water Stress Corrosion Cracking (PWSCC), was repaired using manual grinding and welding. In February 2001, the manual repair of several CRDM nozzles at Oconee Unit 3 with similar defects resulted in extensive radiation dose to the personnel due to the location and access limitations. Consequently, the B&W Owner's Group (BWOG) commissioned Framatome ANP (FRA-ANP) to design and demonstrate an automated repair that was ultimately implemented at Oconee Unit 2.

Due to concerns that similar CRDM nozzle degradation may have occurred at other Pressurized Water Reactors (PWRs), Florida Power & Light (FP&L) has contracted FRA-ANP to adapt this repair for its Turkey Point Units 3 & 4(TP-3 & 4) with modifications as required to meet ASME Code Case N-638.

3.0 Finite Element Model

The model used in this analysis is an ANSYS model (of the original design) documented in Ref. 1 and modified here to reflect the changes due to the temperbead weld repair procedure. It is a 3-dimensional model of a 180 degree segment of a CRDM tube with the adjacent head region and interconnecting weld. Symmetry boundary conditions are used to represent the un-modeled portions. The model is shown in Fig. 3.1. The dimensions and material properties of the original design are also documented in Ref. 1.

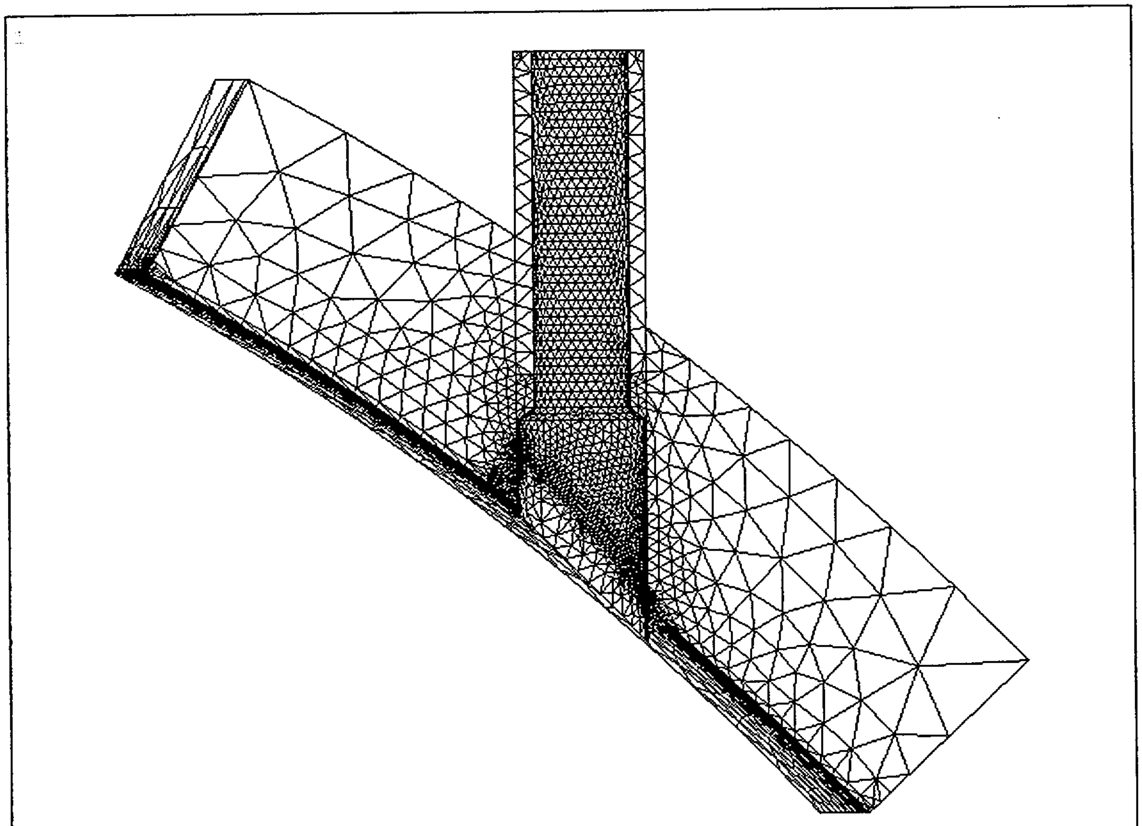

Fig. 3.1 Finite Element Mesh

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4.0 Material Properties

The material properties of the original design are documented in Ref. 1. The material properties for the repair weld are listed in Table 4.1.

Table 4.1 - Repair Weld Material Properties									
ALLOY 690									
TEMP	E	μ	ρ	α	k	C	Sm	Sy	Su
100	30.1	0.3	0.3060	7.76	0.5833	0.1034	23.3	35.0	80.0
200	29.5	0.3	0.3053	7.85	0.6333	0.1075	23.3	31.6	80.0
300	29.1	0.3	0.3045	7.93	0.6833	0.1113	23.3	29.8	80.0
400	28.8	0.3	0.3038	8.02	0.7333	0.1140	23.3	28.7	80.0
500	28.3	0.3	0.3030	8.09	0.7833	0.1173	23.3	27.8	80.0
600	28.1	0.3	0.3023	8.16	0.8333	0.1189	23.3	27.6	80.0
700	27.6	0.3	0.3016	8.25	0.8833	0.1218	23.3	27.6	80.0
Ref.	5,6	Assumed	5,6	5,6	5,6	Calc.	5,6	5,6	5,6

5.0 Model Boundary Condition

The outer surfaces of the RV Head thickness are assigned thermal boundary conditions that are insulated (adiabatic). Structurally they are allowed only to deflect in the direction that is radial to the head center of curvature.

For thermal transient type loads (heat transfer coefficient and bulk fluid temperature), the appropriate surfaces are loaded. For the interface between the Primary coolant water temperature and the cladding/J-groove weld (i.e., inside the reactor vessel head), a heat transfer coefficient associated with a 'turbulent' condition is applied. Per Reference 7, a film coefficient of [ ] Btu/hr-ft^2-F is used in this analysis. For the inside diameter of the CRDM Housing nozzle, the same heat transfer coefficient for the inside head is applied even though it is expected that there is lack of forced flow due to much limited space. At the RV Head exterior surface, a relatively small film coefficient (representing heat loss through the insulation) is applied in conjunction with the estimated ambient temperature above the head. The small gap between the remaining CRDM Housing nozzle OD and penetration bore are modeled as 'coupled temperatures' to best represent the actual condition.

For pressure, those surfaces in contact with primary coolant water are loaded. These include the RV Head/J-groove weld, CRDM Housing nozzle internal extension and inside diameter. The exterior of the RV Head (and the interface gap between the CRDM Housing nozzle and penetration bore) are not loaded by pressure. The upper end of the CRDM Housing nozzle cylinder has a pressure applied to represent the hydrostatic end load from the CRDM closure.
A portion of the remaining CRDM Housing nozzle is roll-expanded to the wall of the adjacent penetration bore (see Reference 2). This roll-expansion fit limits the relative motions of the

CRDM nozzle body and the RV Head as the shrink fit (i.e., interference fit) did for the original model. By limiting the relative motions, the thermal and pressure induced stresses in the interconnecting temperbead weld are reduced. The shrink fit effect is demonstrated analytically by comparing the results of runs 'LWDesign2.out' (w/ interference restraint) and 'LWDesign3.out (w/o interference restraint) from Ref. 13. To assure conservative results, no credit is taken for this effect in the model – the restraint provided by the roll-expansion is omitted.

The model is subjected to the Reactor Coolant outlet thermal and pressure conditions versus time. Per Reference 7, the thermal transients are grouped in 3 cases: Heat-up/Cool-down, Plant loading/unloading, and bounding remaining transients.

Table 5.1 Transients

Case	Transients	Cycles
HUCD	HUCD (200 cycles) Hydrotest* (5 cycles)	205
Plant Loading/Unloading	Plant Loading/Unloading	14500
Remaining	10% Step Increase (2000 cycles) 10% Step Decrease (2000 cycles) Large Step Decrease (200 cycles) Loss-of-Load (80 cycles) Loss-of-Flow (80 cycles) Reactor trip (400 cycles) Loss-of-AC Power (40 cycles)	2800

Note: * Hydrotest includes 2500 psia @ operating temp and 1 cycle of 3125 psia @ 100 °F.

The temperature and pressure values assumed for the above transients are shown in the following pages.

Table 5.2 HUCD Transient

HUCD		
Time (hrs)	Temperature (°F)	Pressure (psi)
0	100	300
2	300	300
5	600	2235
11	600	2235
14	300	300
16	100	300
19	100	300

Table 5.3 PL_LU Transient

Plant Loading/Unloading		
Time (hrs)	Temperature (°F)	Pressure (psi)
0	547	2235
0.3333	618	2235
3	618	2235
3.3333	547	2235
5	547	2235

Table 5.4 Remaining Transients

Remaining Transients		
Time (hrs)	Temperature (°F)	Pressure (psi)
0	578	2235
0.0028	618	2585
3	618	2585
3.0028	518	2235
6	518	1735

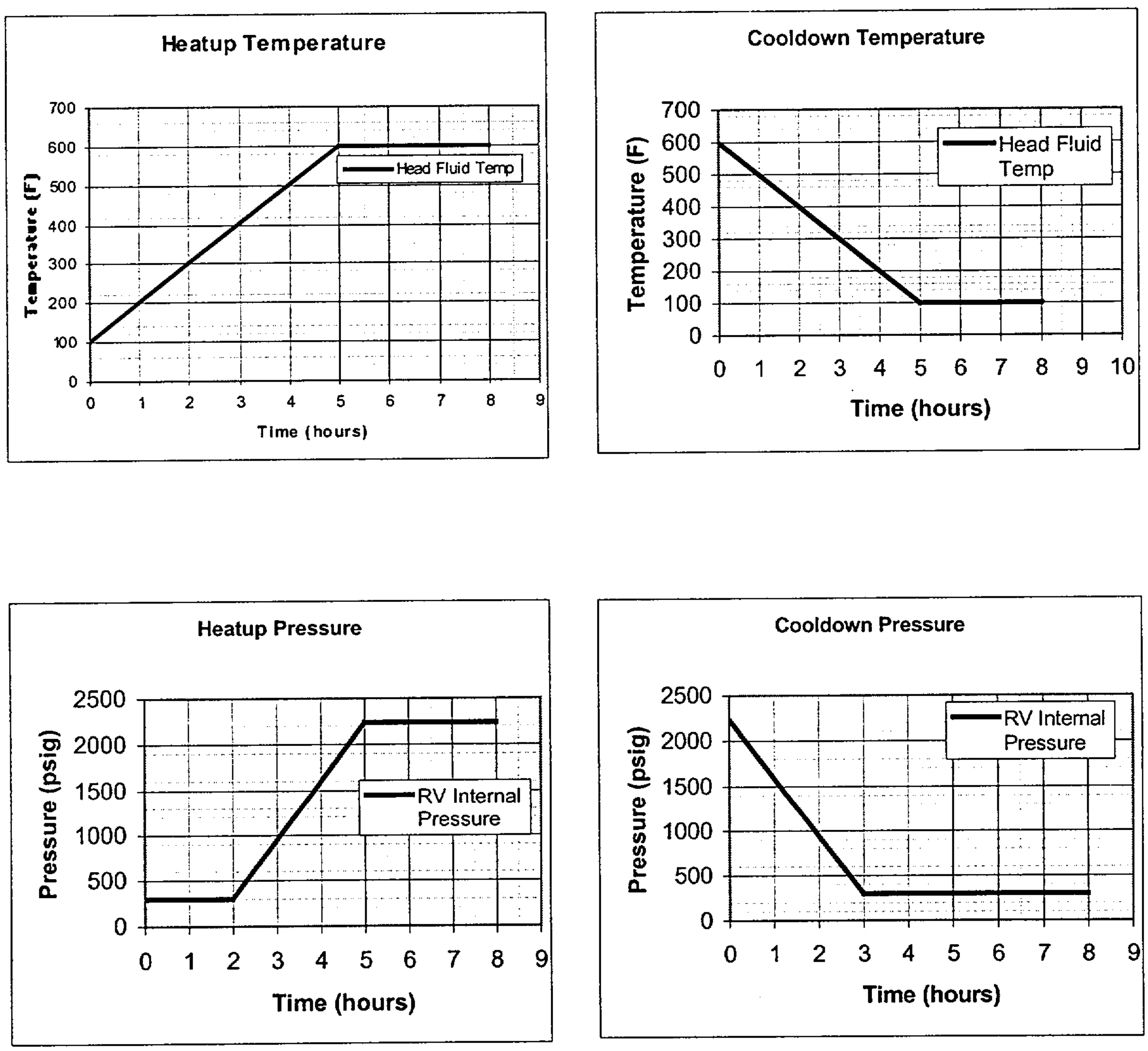

Figure 5.1 HUCD Transient

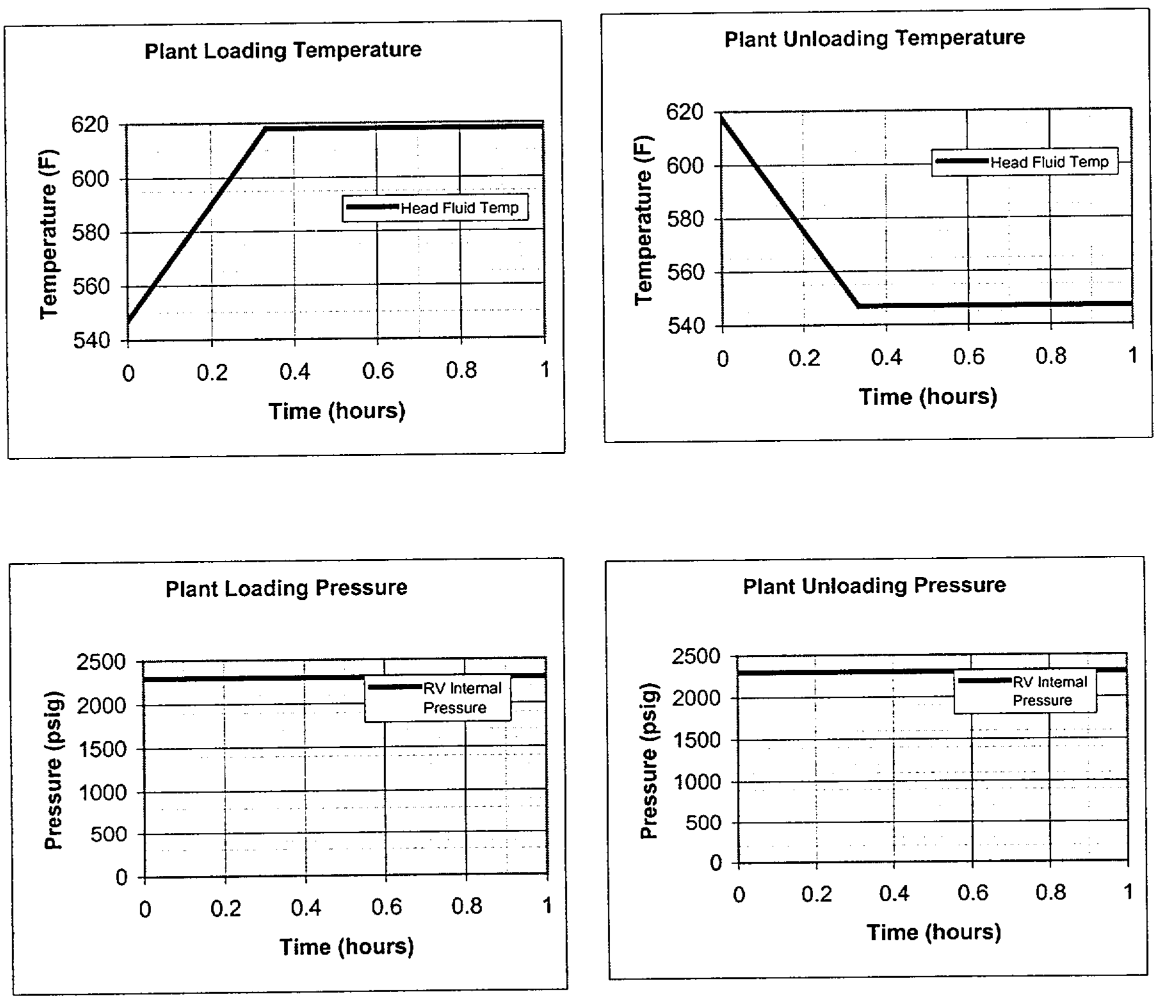

Figure 5.2 Plant Loading/Unloading

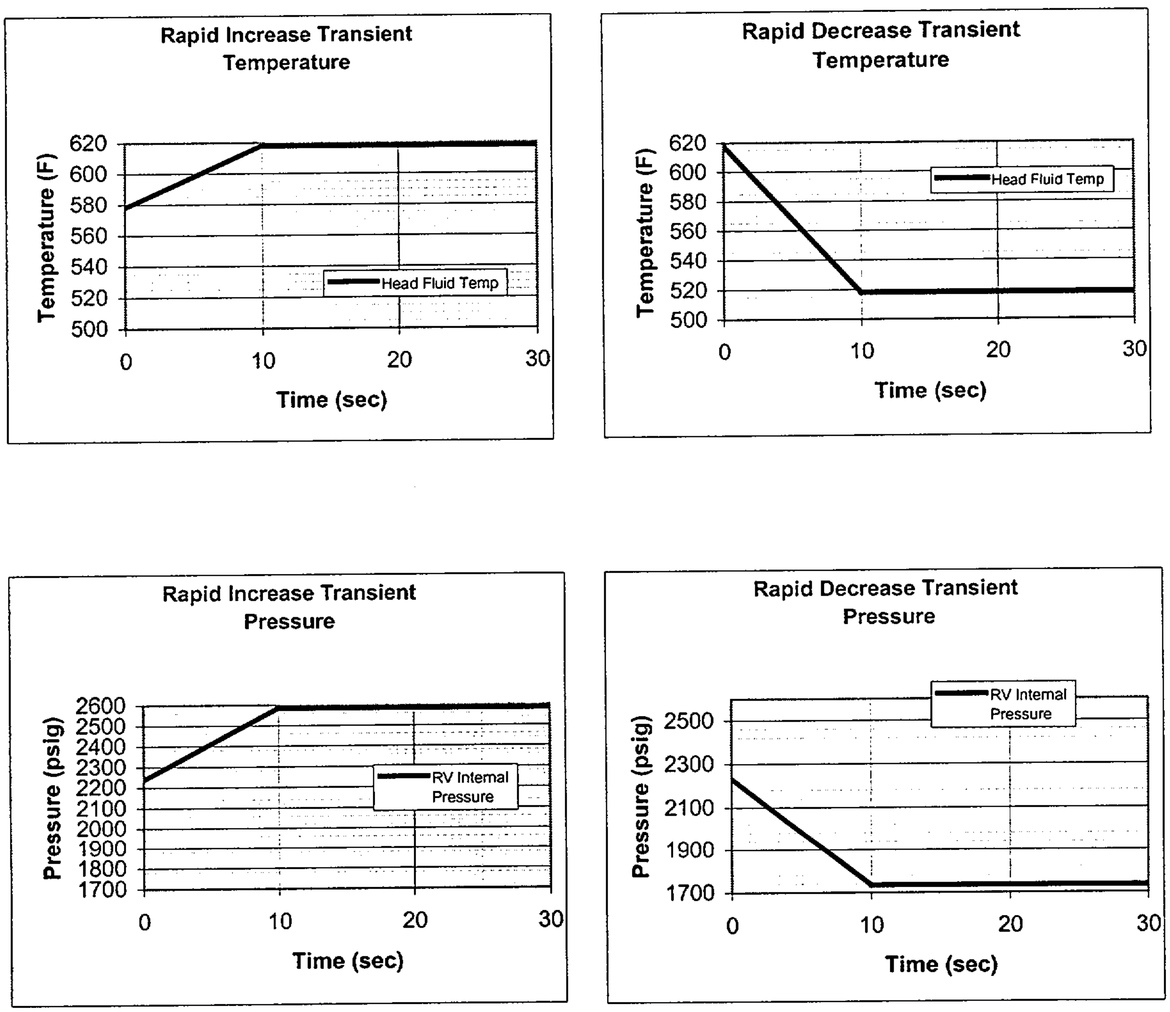

Figure 5.3 Bounding Remaining Transients

6.0 Thermal Results

The results of the heat transfer analysis are contained in the output file ***th.out. The relevant transient results are summarized in the graphs in Fig. 6.1 and Fig. 6.2 (See next pages). These figures depict the temperature versus time and temperature difference versus time. The numerical data in these graphs is in file ***_DeltaTs.out.

Figure deleted for proprietary reason

a) HUCD

Figure deleted for proprietary reason

b) Plant Loading/Unloading

Figure deleted for proprietary reason

c) Rapid Transients

Fig. 6.1 Temp. Plots for Three Transient Groups

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Figure deleted for proprietary reason

a) HUCD

Figure deleted for proprietary reason

b) Plant Loading/Unloading

Figure deleted for proprietary reason

c) Rapid Transients

Fig. 6.2 Delta-T Plots for Three Transient Groups

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Note) 2 is a node on cladding, 5 is a node on middle of the closure head, and 7 is a node on outside of the closure head.

Based on the delta-T values depicted above (and steady-state conditions), stress calculations were done at the following time points in the transients:

Table 6.1 Load cases for Static Runs

HUCD Transient		
Load case	Time (hr)	Description
1	0.001	Initial condition
2	5.0	End of Heatup
3	11.0	End of Steady State
4	13.124	Max. Delta T
5	14.0	Pressure drops to 300 psi
6	16.0	End of Cooldown
Plant Loading/Unloading Transients		
Load case	TIME(Hr)	Description
1	0.001	Initial condition
2	0.333	End of Plant Loading
3	3.000	End of Steady State
4	3.333	End of Plant Unloading
Remaining Transients		
Load case	TIME(Hr)	Description
1	0.001	Initial condition
2	0.002778	End of Rapid Heatup
3	0.13694	Slightly after end of Rapid Heatup
4	3.0	End of Steady State
5	3.002778	End of Rapid Cooldown
6	3.406	Slightly after end of Rapid Cooldown

* Transient time scale is as defined in Reference 1.

7.0 Stress Results

Stress analysis is performed at each of the previously listed time points. The model is loaded by nodal temperatures (thermal gradients) and internal pressure (see Table 6.1 for applicable values). The results of the stress analyses are contained in the output file ****st.out. The ANSYS (Ref. 3) post-processor was used to tabulate the stresses along paths through the weld and head and classify them in accordance with ASME Code criteria. The location of the paths are shown in Figures 7.1 and 7.2. A review of the stress results indicates that these paths include the highest stressed (limiting) locations for the assembly (including RV head, CRDMH nozzle, connecting repair weld and remnants of the original weld).

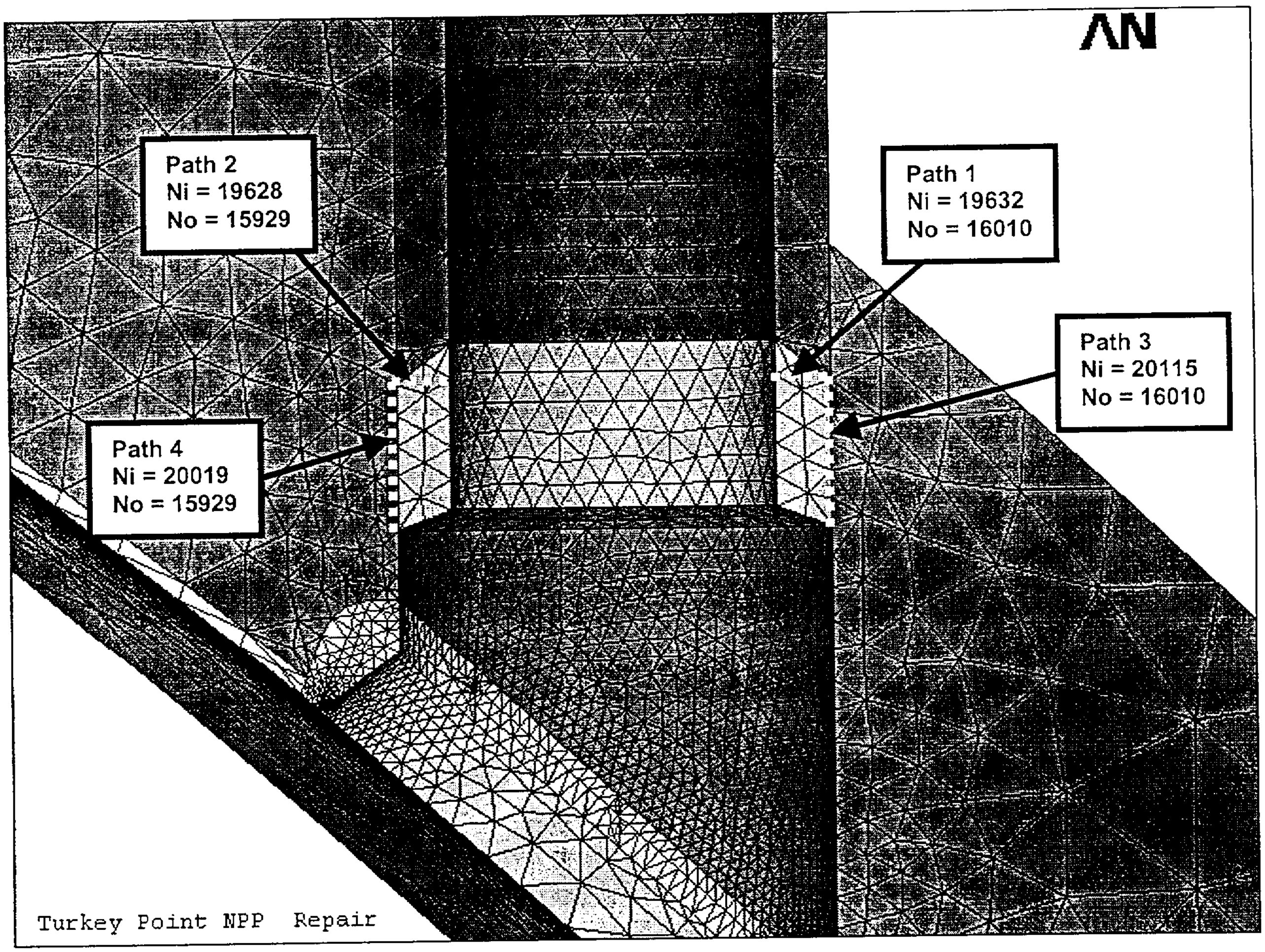

Fig. 7.1 Stress Paths Through Weld

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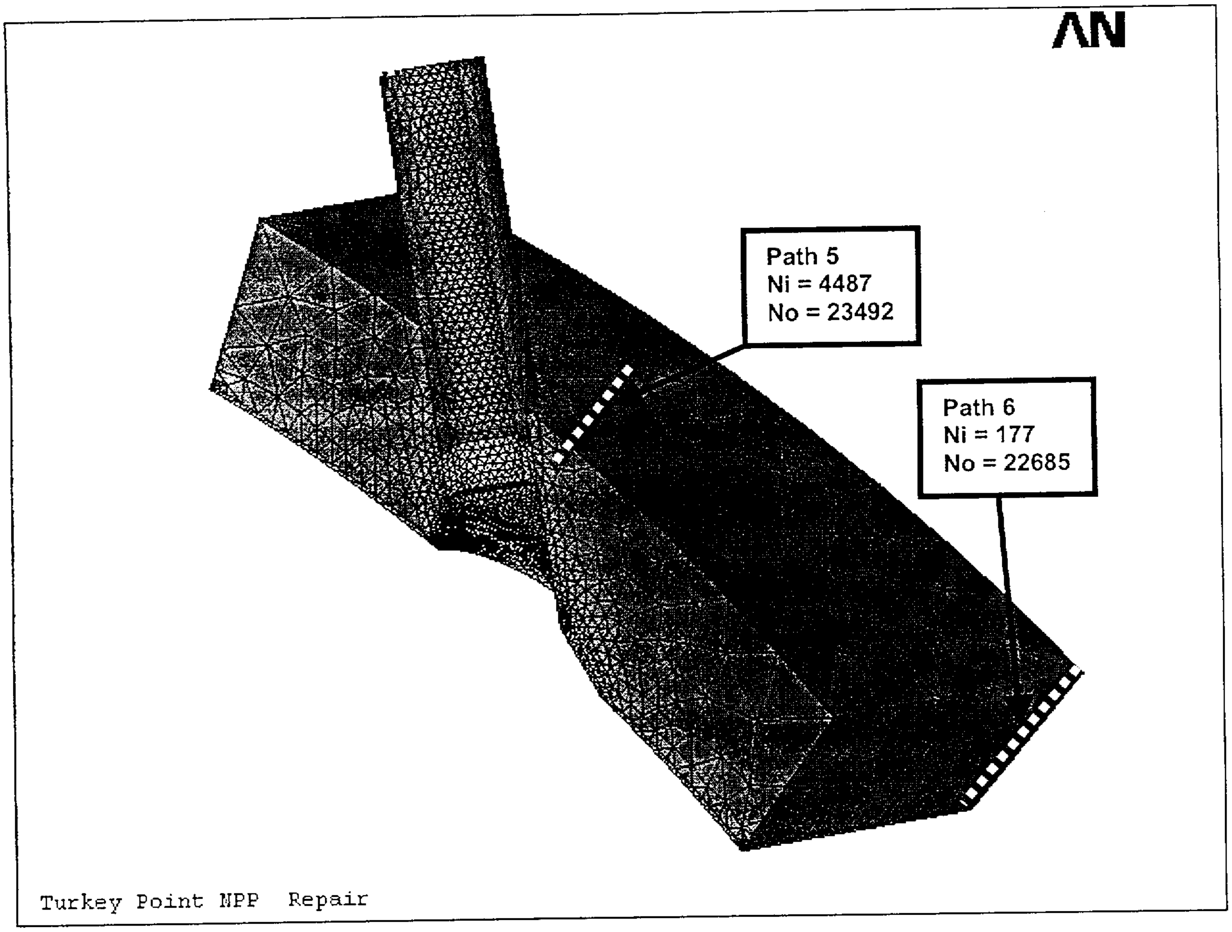

Fig. 7.2 Stress Paths Through Head

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The results from the stress classification post-processing run are contained in the file ***_weld_path.out. This run calculated the classified stress components (membrane, bending and peak) for each of the stress paths shown in Figs. 7.1 and 7.2, at each of the time points analyzed in the stress analysis. Another post-processing program, contained in file SummaryForm.frm, uses the data from ***_path _weld.out to calculate stress intensity ranges for use in fatigue calculations, following the method prescribed by the ASME Code in Paragraph NB-3216.2. The cycles associated with the calculated stresses are defined in Reference 1.

The stresses resulting from the thermal/pressure transients represent the dominant contribution to total stresses for the repaired configuration of the RV Head, CRDMH Nozzle and connecting repair weld. It is acknowledged that there are mechanical loads applied at the CRDMH Nozzle flanged connection (outboard of the RV Head) and some load from the bolting-up of the RV Head Closure.

The CRDM Housing nozzles function as mechanical mounts for the Control Rod Drive Mechanisms. The Control Rod Drive Mechanisms are relatively tall slender structures that may be subjected to loads from seismic or other motions. Any movement of the Control Rod Drive Mechanisms produce loads in the CRDM Housing nozzles (essentially cantilevered from the RV Head). However, the design of the CRDM Housing nozzle connection to the RV Head includes a roll-expansion fit feature. This fit is located above the 'CRDM Housing-to-RV Head connection' weld. Therefore, the mechanical loads from the Control Rod Drive Mechanisms are transmitted to the RV Head through the roll-expansion fit region. This design feature effectively shields the 'CRDM Housing-to-RV Head connection' repair weld from being subjected to external mechanical loads. Therefore, no external mechanical loads are applicable to the analysis of the 'CRDM Housing-to-RV Head connection' repair weld. [This approach is consistent with the original stress report (Ref. 9). Thus, the mechanical load stresses are unchanged for the CRDMH Nozzles.]

As for boltup load, it is assumed that the load is negligible. Furthermore, the boltup load will not be used in Fatigue Analysis because it is constant load during operating conditions. Therefore, Closure Head boltup load is considered to be insignificant with regard to the overall stress levels resulting from other loadings.

7.1 ASME Code Criteria

The ASME Code stress analysis involves two basic sets of criteria – 1) assure that failure does not occur due to a single application of the design mechanical loads and 2) assure that failure does not occur due to repetitive loadings.

In general, the Primary Stress Intensity criteria of the ASME Code (Ref. 4) demonstrates that the design is adequate for a single application of design mechanical loads.

Also, the ASME Code criteria for cumulative fatigue usage factor assures that the design is adequate for repetitive loadings.

7.2 ASME Code Primary Stress (SI) Intensity Criteria

The analysis of primary stress intensities for Design Conditions is made to satisfy the requirements for single application of design loads in accordance with Reference 4, par. NB-3221.

Other related criteria include the design limits for minimum required pressure thickness (see NB-3324) and reinforcement area (see NB-3330). The requirements for minimum required pressure thickness are effectively addressed by meeting the Primary General Membrane Stress Intensity criterion as shown below. Also, the 'reinforcement area requirements' are superceded by demonstrating that all of the stress requirements have been met. This approach is permitted as stated in par. NB-3331(c) of Reference 4.

7.2.1 Primary Stress Intensities for Design Conditions (Design Pressure @ Design Temperature)

Per Reference 7, Design Pressure = 2500 psig; Design Temperature = 650F

Computer run "TP_pres.out" contains the stress solution for the design conditions. The post-processing run "pres_path_weld.out" contains the classification of stresses into categories that are comparable to the categories used in the criteria of the ASME Code as discussed below:

NB-3221.1 – General Primary Membrane Stress Intensity (Pm ≤ 1.0 Sm)

The applicable value occurs remote from discontinuities and includes no local effects. From Figure 7.2, Path 6 depicts an appropriate location for the RV Head. From "pres_path_weld.out", Path 6, the membrane stress intensity = [ ] ksi. For the RV Head material, Sm = **26.7** ksi (Ref. 1, Section 3.2). Therefore, the requirement is met for the RV Head.

For the CRDMH Nozzle (the portion affected by the repair), the membrane stress intensity is maximum at the thinned (by remediation) section (see Ref. 2). This value is calculated as Pm = ((Pr/t) – (P/2)) = ((2.5 ksi*1.497"/0.503") – (2.5 ksi/2)) = **6.2** ksi. This is less than 1.0 Sm for SB-167 (Alloy 600) = **23.3** ksi (Per Ref. 1). Therefore, the requirement is met for the CRDMH Nozzle wall (as well as the corresponding section of the A690 weld).

NB-3221.2 – Local Membrane Stress Intensity (Pl $\leq$ 1.5 Sm)

The applicable value includes the effect of discontinuities and includes no stress concentration effects (such as in the near vicinity of the penetration bore). From Figure 7.2, Path 5 depicts an appropriate location for the RV Head. This location is at a distance equivalent to midway between two CRDMH Nozzle penetrations. The local effect (i.e., the amount above 'general membrane') is doubled to superimpose the effect of the un-modeled adjacent CRDMH Nozzle. From "pres_path_weld.out", Path 5, the membrane stress intensity = [ ] ksi. Therefore, the local effect = [ ] ksi – [ ] ksi = [ ] ksi. Thus, considering two adjacent CRDMH Nozzles, the Primary Local Membrane SI for the RV Head material = [ ] ksi + [ ] ksi = [ ] ksi. For the RV Head material, 1.5 Sm = **40.1** ksi (Ref. 1, Section 3.2). Therefore, the requirement is met for the RV Head.

For the CRDMH Nozzle wall section, the membrane SI values at the lower end (at the elevation of the crevice bottom) are classified as 'secondary' per NB-3337.3(b) of Reference 4. This 'secondary stress' classification is dependent on the weld dimensions fulfilling the requirements of Figure NB-4244(d)-1 and par. NB-3352.4(d). Figure 7.3 herein depicts the designer's concept of the repair weld being enveloping the Code required weld. It is concluded, then, that the repair weld is larger (and stronger) than the minimum required by the Code. Thus, there are no loads that generate Primary Local Membrane SI in the CRDMH Nozzle wall. Therefore, for the CRDMH Nozzle wall – Pl includes the Pm contribution; therefore, Pl = [ ] ksi $\leq$ 1.5 Sm = **35.0** ksi for SB-167 & A690 and the requirement is met.

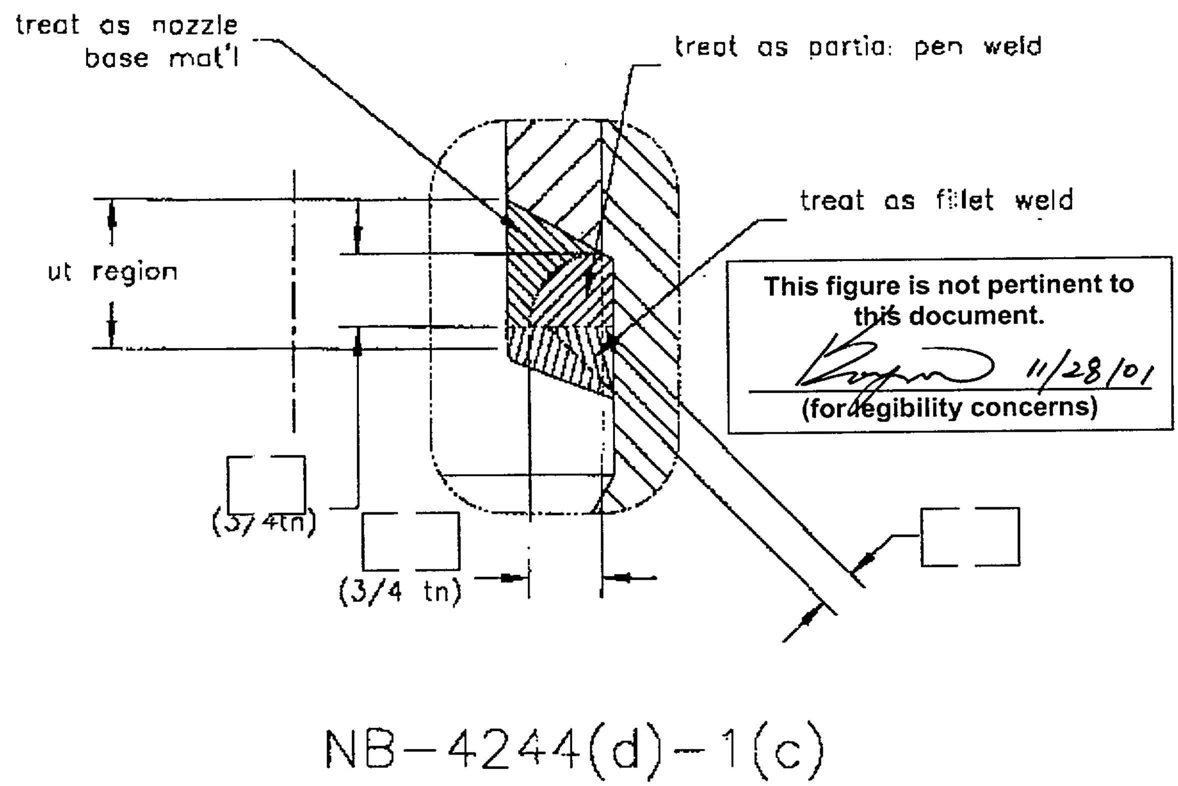

FIGURE 7.3

NB-3221.3 – Primary Membrane + Primary Bending SI (Pl+Pb ≤ 1.5 Sm)

The applicable value includes the effect of discontinuities and includes no stress concentration effects (such as in the near vicinity of the penetration bore). From Figure 7.2, Path 5 depicts an appropriate location for the RV Head. This location is at a distance equivalent to midway between two CRDMH Nozzle penetrations. The local effect (i.e., the amount above 'general membrane') is doubled to superimpose the effect of the un-modeled adjacent CRDMH Nozzle. From "pres_path_weld.out", Path 5, the 'membrane+bending' stress intensity = [ ] ksi. Therefore, the local 'membrane+bending' effect = [ ] ksi – [ ] ksi = [ ] ksi. Thus, considering two adjacent CRDMH Nozzles, the Primary Membrane + Primary Bending SI for the RV Head material = [ ] ksi + [ ] ksi = [ ] ksi. For the RV Head material, 1.5 Sm = **40.1** ksi (Ref. 1, Section 3.2). Therefore, the requirement is met for the RV Head.

Per Ref. 4, Table NB-3217-1, there is no Primary Bending stress in the CRDMH Nozzle. Therefore, Pl+Pb = Pl = [ ] ksi (same as Pm) ≤ 1.5 Sm = **35.0** ksi for SB-167 & A690 and the requirement is met.

7.2.2 Primary Stress Intensities for Emergency (Level C) Conditions

Since the Level C condition is not mentioned in Reference 7, the maximum allowable pressure from the maximum primary stress is compared to the hydrotest pressure, which is the highest pressure can occur during operating conditions. The maximum primary stress is calculated from the service limits. It is then used to obtain the maximum allowable pressure from the ratio of the maximum primary stress to Design Pressure Stress Intensity.

RV Head (max. values considering all regions of low-alloy material):

Note: The repaired configuration generates no 'Primary Bending' stresses in the RV Head

Sm = 26.7 ksi Sy = 43.5 ksi

Max. Primary General Membrane SI = 1.2 Sm = 32 ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3224.1] (A302 Gr. B @650F)

Max. Primary Local Membrane SI = 1.8 Sm = 48.1 ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3224.1] (A302 Gr. B @650F)

Max. Primary Membrane + Primary Bending SI = 1.8 Sm = 48.1 ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3224.1] (A302 Gr. B @650F)

CRDMH Nozzle/Weld (max. values considering all regions of high-alloy material):

Sm = 23.3 ksi Sy = 27.5 ksi

Max. Primary General Membrane SI = Sy = 27.5 ksi
Allowable Pressure = ([ ])*25000 psi = [ ] psi
[Ref. 4, Par. NB-3224.1] (A600/A690 @650F)

Max. Primary Local Membrane SI = 1.5 Sy = 41.25 ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3224.1] (A600/A690 @650F)

Max. Primary Membrane + Primary Bending SI = 1.5 Sy = 41.25 ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3224.1] (A600/A690 @650F)

Comparing allowable pressures with 3125 psia, which is the Maximum pressure from the hydrotest, it is observed that they are bigger than the hydrotest pressure.
Therefore, as long as the Max. Primary SI is less than the service limit, the structural integrity of the repair weld design is acceptable for the Level C condition.

7.2.3 Primary Stress Intensities for Faulted (Level D) Conditions

Since the Level D condition is not mentioned in Reference 7, the maximum allowable pressure from the maximum primary stress is compared to the hydrotest pressure, which is the highest pressure can occur during operating conditions. The maximum primary stress is calculated from the service limits. It is then used to obtain the maximum allowable pressure from the ratio of the maximum primary stress to Design Pressure Stress Intensity.

<u>RV Head (max. values considering all regions of low-alloy material):</u>
Note: The repaired configuration generates no 'Primary Bending' stresses in the RV Head
Sm = 26.7 ksi   Sy = 43.5 ksi   Su = 80 ksi

Max. Primary General Membrane SI = 0.7 Su = 56 ksi
Allowable Pressure = ([   ])*2500 psi = [   ] psi
[Ref. 4, Par. NB-3225, F-1331.1(a)]   (A302 Gr. B @650F)

Max. Primary Local Membrane SI = 1.05 Su = 84.0 ksi
Allowable Pressure = ([   ])*2500 psi = [   ] psi
[Ref. 4, Par. NB-3225, F-1331.1(b)]   (A302 Gr. B @650F)

Max. Primary Membrane + Primary Bending SI = 1.05 Su = 84.0 ksi
Allowable Pressure = ([   ])*2500 psi = [   ] psi
[Ref. 4, Par. NB-3225, F-1331.1(c)]   (A302 Gr. B @650F)

<u>CRDMH Nozzle/Weld (max. values considering all regions of high-alloy material):</u>
Sm = 23.3 ksi   Su = 80 ksi

Max. Primary General Membrane SI = 2.4 Sm = 55.9 ksi
Allowable Pressure = ([   ])*2500 psi = [   ] psi
[Ref. 4, Par. NB-3225, F-1331.1(a)]   (A600/A690 @650F)

Max. Primary Local Membrane SI = 3.6 Sm = 83.9 ksi
Allowable Pressure = ([   ])*2500 psi = [   ] psi
[Ref. 4, Par. NB-3225, F-1331.1(b)]   (A600/A690 @650F)

Max. Primary Membrane + Primary Bending SI = 3.6 Sm = 83.9 ksi
Allowable Pressure = ([   ])*2500 psi = [   ] psi
[Ref. 4, Par. NB-3225, F-1331.1(c)]   (A600/A690 @650F)

Comparing allowable pressures with 3125 psia, which is the Maximum pressure from the hydrotest, it is observed that they are bigger than the hydrotest pressure.
Therefore, as long as the Max. Primary SI is less than the service limit, the structural integrity of the repair weld design is acceptable for the Level D condition.

7.2.4 Primary Stress Intensities for Test Conditions

Reference 7 specifies only one 'test' condition that is significant to the stress levels in the Closure Head (includes CRDMH Nozzle repair region) – Hydrotest case. This transient results in a pressure of 3125 psia. Thus, the pressure induced Primary Stresses due to these transients are greater than those calculated for the Design Condition.

<u>RV Head (max. values considering all regions of low-alloy material):</u>
Note: The repaired configuration generates no 'Primary Bending' stresses in the RV Head

Max. Primary General Membrane SI = 0.9 Sy = 45.0 ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3226(a)] (A302 Gr. B @100F)

Max. Primary Membrane + Primary Bending SI = 2.15Sy–1.2 Pm = **39.5** ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3226(b)] (A302 Gr. B @100F)

<u>CRDMH Nozzle/Weld (max. values considering all regions of high-alloy material):</u>

Max. Primary General Membrane SI = 0.9 Sy = **31.5** ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3226(a)] (A600/A690 @100F)

Max. Primary Membrane + Primary Bending SI = 2.15Sy–1.2 Pm = **37.5** ksi
Allowable Pressure = ([ ])*2500 psi = [ ] psi
[Ref. 4, Par. NB-3226(b)] (A600/A690 @100F)

Comparing allowable pressures with 3125 psia, which is the Maximum pressure from the hydrotest, it is observed that they are bigger than the hydrotest pressure.
Therefore, as long as the Max. Primary SI is less than the service limit, the structural integrity of the repair weld design is acceptable for the hydrotest condition.

7.3 ASME Code Primary+Secondary SI Range and Fatigue Usage Criteria

As stated previously, the analysis of stresses for transient conditions is required to satisfy the requirements for repetitive (or cyclic) loadings. The following discussion describes the fatigue analysis process employed herein for the Nozzle opening inside repair design.

As described in section 7.0, the stresses for each transient time point chosen for stress analysis are determined in the ANSYS solution run "TP_***_st.out" (for loadings due to thermal gradients and corresponding pressures).

Overall stress levels are reviewed and assessed to determine which model locations require detailed stress/fatigue analysis. The objective is to assure that 1) the most severely stressed locations are evaluated and 2) that the repair region is quantitatively qualified.

Once the specific locations for detailed stress evaluation are established, the ANSYS *'paths'* (sometimes called 'stress classification lines', SCL) are defined. Post-processing runs for these paths are made to convert the raw component stresses along these paths into Stress Intensity (SI) categories that correlate to the criteria of the ASME Code (i.e., 'membrane', 'linearized membrane+bending' & 'total').

The transient analysis of the repair configuration indicates that the location of prime importance is at the bottom of the crevice between the nozzle OD and the penetration bore diameter. This location includes the maximum peak stresses (due to the applicable SCF of 4.0) and includes the low-alloy RV Head base metal (has lower fatigue properties compared to the high-alloy material). To assure that the maximum stress values are obtained, paths are taken through the weld in a radial direction (relative to the nozzle) and through the weld in a vertical direction along the 'weld-to-RV Head' interface. These sectional locations are analyzed at the 'downhill' and 'uphill' side of the model (see Figure 7.1). Review of the stress results and experience with analyses of similar hillside configurations indicate that these sections (4 total) include the location of maximum stress/usage. The stress linearization for these paths (1 – 4) are contained in computer file "***_path _weld.out".

However, because this is a 3-D analysis and the directions of the principal stresses may vary during the transient, the 'range' of 'linearized membrane+bending' is determined by the method prescribed by Paragraph NB-3216.2 of the ASME Code (Ref. 4). The computer runs containing the results of the application of this method are "****_path_weld.Class_Line_Summary". The maximum range value as determined in these runs are compared directly to the Primary + Secondary Stress Intensity Range criteria of the ASME Code.

For consideration of fatigue usage, the 'Peak Stress Intensity Ranges' are calculated. These values must include the 'total' localized stresses. As mentioned above, the geometry of the repair design results in a crevice-like configuration between the nozzle OD and the penetration bore diameter. Therefore, the 'linearized membrane+bending' stress intensity range at this location (Paths 1-4, outside) is multiplied by a factor of 4.0 (Ref. 4, Par. NB-3352.4(d)(5)) to represent the 'Peak Stress Intensity Range'. *[Note: The resulting values are* *confirmed to be greater than the 'total' stress intensities calculated directly from the model.]*

As documented in Reference 7, the transients that have a potential impact on fatigue usage are divided into three groups – HUCD, Plant Loading/Unloading Transients and Rapid Transients. The associated cycles (based on a 40 year plant life) for these transients are:

HUCD = 200 cycles

Plant Loading/Unloading = 14500 cycles

Rapid Transient = 2800 cycles

For conservative approach, RV Head Base metal is looked at for Fatigue Usage Factor calculation because of lower fatigue properties as opposed to high alloy material. Also, instead of checking all the nodes from the said Stress Class Lines, the biggest range of SI is chosen from each transients and is used for the Fatigue Usage Factor calculation.

<u>Maximum Primary + Secondary SI Range for Low-alloy Material in HUCD</u>

Ref. Run "HUCD_weld_path.Class_Line_Summary"

Max. P+S SI Range = [ ] ksi (Path [ ], inside)

This is less than the maximum allowed by the design code (3 Sm = **80.0** ksi)

Note) Path 2-inside has 45.7 ksi. However, this is not used because it is not multiplied by factor of 4

<u>Maximum Primary + Secondary SI Range for Low-alloy Material in PL_LU</u>

Ref. Run "PL_LU_weld_path.Class_Line_Summary"

Max. P+S SI Range = [ ] ksi (Path [ ], inside)

This is less than the maximum allowed by the design code (3 Sm = **80.0** ksi)

<u>Maximum Primary + Secondary SI Range for Low-alloy Material in RA</u>

Ref. Run "RA-weld_path.Class_Line_Summary"

Max. P+S SI Range = [ ] ksi (Path [ ], outside)

This is less than the maximum allowed by the design code (3 Sm = **80.0** ksi).

Using the ranges/cycles described above, the corresponding cumulative usage is calculated on the following pages.

EVALUATION TITLE: *TP - CRDM Temperbead Weld Analysis*

LOCATION: *Intersection of Pad bottom surface, Sleeve OD and RV Head penetration bore (crevice region; FSRF = 4.0)*
*(Ref. Run "*_weld_path.Class_Line_Summary")*

MATERIAL: *A302 Gr. B* *(both high alloy and low-alloy steels are present at this crevice region)*
TYPE: *Low-alloy steel*
UTS (psi) = *80000*
E matl (psi) = *2.64E+07* *(at T = 600F)* **E ratio =** *('E curve' / 'E analysis')*

	COND. NUMBER	TRANSIENTS WITH RANGE EXTREMES	REQ'D CYCLES 14 years	PEAK SI RANGE	E matl	S alt	(E ratio) x S alt	ALLOWABLE CYCLES "N"	USAGE FACTOR "U"
RV Head	1	HU/CD/Hydro	72	[ ]	2.64E+07	[ ]	[ ]	[ ]	[ ]
RV Head	2	PL_LU	5003	[ ]	2.64E+07	[ ]	[ ]	[ ]	[ ]
RV Head	3	RA	980	[ ]	2.64E+07	[ ]	[ ]	[ ]	[ ]

Total Low-alloy Usage = *[ ]*

Note: The 'Peak SI Range' = 'Linearized Membrane + Bending' x Fatigue Strength Reduction Factor (FSRF)

For cycle group 1, 'Linearized Memb + Bending' SI Range = [ ] ksi; FSRF = 4.0

For cycle group 2, 'Linearized Memb + Bending' SI Range = [ ] ksi; FSRF = 4.0

For cycle group 3, 'Linearized Memb + Bending' SI Range = [ ] ksi; FSRF = 4.0

8.0 Consideration of Corrosion of RV Head Low-Alloy Material

The design configuration of the CRDMH Nozzle Temperbead repair results in an area of RV Head base material (low alloy; SA302 Gr. B) being exposed to continuous contact with Reactor Coolant water. The chemistry of the Reactor Coolant combined with the properties of the RV Head material result in corrosion of the wetted surface.

The amount of corrosion rate has been determined to be [ ] inch per year (Reference 12). At this rate, the total surface corrosion for a repair life of [ ] years of plant life (Reference 7) is [ ] inch. This represents the maximum increase in bore radius for the operating period.

The significance of the increased bore diameter (of this magnitude) is acceptable based on the rational included in Appendix A of Reference 13.

In conclusion, the corrosion of the exposed low-alloy material has a negligible impact on the thermal/structural response of the CRDMH Nozzle assembly with temperbead repair and is, therefore, acceptable.

9.0 Conclusions

The preceding calculations demonstrate that the CRDMH Nozzle temperbead repair design meets the stress and fatigue requirements of the Design Code (ASME Code, Section III, 1989 edition w/o addendum – Reference 4).

The conservative fatigue analysis indicates that the repair design is acceptable for at least [ ] years of operation.

Since the fatigue is a 'linear function' of the cycles, the qualified operating life is [ ] years/[ ] = [ ] years. This result is conservative and could be improved (qualified for more years) by refined determination of the Fatigue Strength Reduction Factor at the intersection of the lower edge of the repair weld with the penetration bore.

10.0 References

1. FRA-ANP Doc. 32-5014129-00, "Turkey Point - CRDMH Connection 3D FE Model"
2. FRA-ANP Dwg. 02-5014781E-00, "CRDM Nozzle ID Temperbead Weld Repair"
3. "ANSYS" Finite Element Computer Code, Version 5.7, Swanson Analysis Systems, Inc., Houston, Pa.
4. ASME Boiler and Pressure Vessel Code, Section III, 1989 with no addenda.
5. ASME Boiler and Pressure Vessel Code, Code Case N-474, Design Stress Intensities and Yield Strength Values for Alloy 690 with a Minimum Yield Strength of 35 ksi - Class 1 Components, Section III, Division 1, approved March 5, 1990.
6. FRA-ANP Doc. 51-1176533-00, "Alloy 690 Material Properties"
7. FRA-ANP Doc. 51-5014575-00, "TURKEY POINT CRDM NOZZLE ID TEMPER BEAD WELD REPAIR REQUIREMENTS"
8. Not used
9. "Stress Report for Reactor", Design Analysis No. 8, "Control Rod Drive Mechanism Housing", BW Contract No. 620-0116-51/52, FRA-ANP Microfilm Roll No. 80-80,81
10. Not used
11. Not used
12. FRA-ANP Document 51-5012576-00, "Corrosion Evaluation of RV Head Penetration Repair"
13. FRA-ANP Document 32-5012424-01, "CRDM Temperbead Bore Weld Analysis"

11.0 Computer Files

The finite element analyses done in this calculation were made using the ANSYS computer program (Ref. 3). Test cases verifying the suitability and accuracy of this program for this analysis were analyzed and the results of that analysis are included in files VM96.OUT and VM187.OUT.

Computer Output Files

File Name	Description
TP_HUCD_th.out	HUCD thermal transient heat transfer analysis
TP_PL_LU_th.out	PL_LU thermal transient heat transfer analysis
TP_RA_th.out	RA thermal transient heat transfer analysis
TP_HUCD_st.out	HUCD stress analysis
TP_PL_LU_st.out	PL_LU stress analysis
TP_RA_st.out	RA stress analysis
HUCD_DeltaTs.out	HUCD thermal post-processing
PL_LU_ DeltaTs.out	PL_LU thermal post-processing
RA_ DeltaTs.out	RA thermal post-processing
HUCD_path _weld.out	HUCD head/weld stress post-processing
PL_LU_path _weld.out	PL_LU head/weld stress post-processing
RA_path _weld.out	RA head/weld stress post-processing
HUCD_path _weld.Class_Line_Summary	HUCD head/weld stress range tabulation
PL_LU_path _weld.Class_Line_Summary	PL_LU head/weld stress range tabulation
RA_path _weld.Class_Line_Summary	RA head/weld stress range tabulation
TP_pres_out	Design Pressure at Design temp analysis
pres_path _weld.out	Design Press stress classification
VM96.out	Verification case for heat transfer analysis
VM187.out	Verification case for stress analysis

APPENDIX A

Stresses used for Crack Growth Assessments

Purpose

The purpose of this appendix is to provide supplemental stress results of the transient analysis for flaw growth assessments. Two areas are selected for this study: original J-groove weld and new temperbead weld (See Fig.7.1). The original J-groove locations include paths through the remnant portion of the original J-groove welds and adjacent RV head base metal in planes oriented at 45 degree increments around the CRDM opening bore (See Fig. A-1). The stresses tabulated herein are to be used as input to flaw growth assessments.

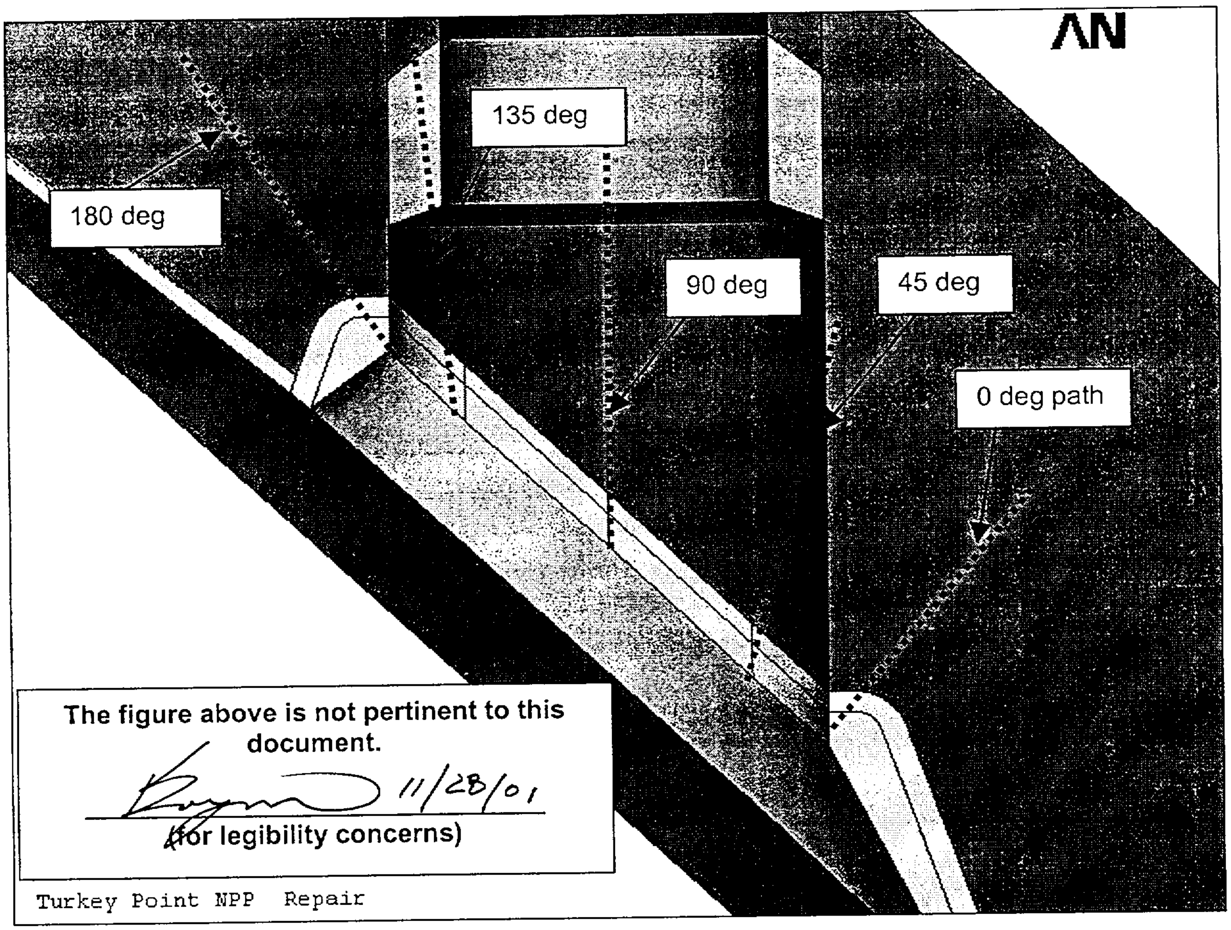

Fig. A-1 Close-up of Paths Through Original Welds/Head

For J-groove weld, there are two line segments in a path: 1) from corner of chamfer to buttering and 2) from buttering to the middle of head thickness. And, each segment has five checking points.

The stress results are in cylindrical coordinate system.
SX = radial to CRDMH Nozzle; SY = hoop; SZ = axial

THE COMPUTER OUTPUT CONTAINING DETAILED STRESSES HAS BEEN REMOVED FOR PROPRIETARY REASONS.

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Computer Files

The ANSYS computer files used for this appendix are following:

Computer Output Files

File Name	Description
A) Original J-groove Weld	
TP_REP_HUCD_path_w1.out	HUCD head/weld stress post-processing
TP_REP_HUCD_path_w2.out	HUCD head/weld stress post-processing
TP_REP_PL_LU_path_w1.out	PL_LU head/weld stress post-processing
TP_REP_PL_LU_path_w2.out	PL_LU head/weld stress post-processing
TP_REP_RA_path_w1.out	RA head/weld stress post-processing
TP_REP_RA_path_w2.out	RA head/weld stress post-processing
B) Temperbead Repair Weld	
TP_REP_HUCD_path_weld_frac.out	HUCD head/weld stress post-processing
TP_REP_PL_LU_path_weld_frac.out	PL_LU head/weld stress post-processing
TP_REP_RA_path_weld_frac.out	RA head/weld stress post-processing