NIAGARA MOHAWK POWER CORPORATION

NUCLEAR ENGINEERING REPORT

NINE MILE POINT 1

SHROUD CRACKING SAFETY ASSESSMENT

FOR

GENERIC LETTER 94-03

REPORT NO. NER-1M-014

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Executive Summary

The Nine Mile Point 1 safety assessment utilizes the BWR Vessel Internals Project (BWRVIP) generic assessment and reviews any differences in shroud cracking susceptibility including: available Nine Mile Point 1 shroud inspections, shroud fabrication, water chemistry, shroud material carbon content, neutron fluence and on-line years specific to Nine Mile Point 1. The generic BWRVIP assessment establishes that the likelihood of up to 360 degree cracking at some depth is fairly high for the H1 through H7 shroud welds. The structural margin assessment, assuming 360 degree cracking, determines that it is unlikely that any cracks exceed 90% depth at the H1 through H7 locations. The visual inspections available for H1,H2,H3,H4 and H7 support this conclusion. The overall structural margin assessment considers the uncertainties in the degree of cracking and recommends that the Oyster Creek shroud inspection results, scheduled to be completed October 1994, be used to assess the uncertainty in estimating the potential shroud cracking depth for the H1 through H7 welds.

The Nine Mile Point 1 specific safety assessment reviews in detail the shroud ring to inconel cone weld (H8) for Nine Mile Point 1 since failure of this weld had been postulated to potentially degrade the core spray system during the design basis recirculation line LOCA. This analysis demonstrates that intergranular stress corrosion Cracking (IGSCC) cracking along the H8 weld is extremely unlikely. The ASME XI inspections of the H8 weld performed during the 1993 refuel outage, were conducted in the areas where this cracking is predicted to initiate. Surface cracking was not observed. Therefore, failure of the H8 weld prior to February 1995 (i.e., 360 degree through-wall cracks) is not considered credible.

The ability of the plant safety features to perform their design basis functions, assuming 360 degree through-wall cracks, is reviewed with the conclusion that for the limiting main steam line break and recirculation line break, control rod insertion is not expected to be impacted and core spray would perform its design basis function. These conclusions are based on preliminary analyses regarding the recirculation asymmetric loads and preliminary detailed main steam line break analyses for which BWRVIP assessment committee detailed analyses are in progress. In addition, a probabilistic safety assessment considering the probability of a design basis event coupled with shroud weld failure is performed. This assessment determines that the overall incremental core damage and large/early release frequency is less than 8.3 E-8 per year or approximately 4 E-8 for a period between now and the scheduled February 1995 refueling outage.

The Nine Mile Point 1 safety assessment supports continued operation until the scheduled February 1995 refuel outage based on the extremely low probability that the core shroud would fail to meet its design basis structural integrity margin during this time period. This, coupled with the extremely low overall probabilistic risk estimate, supports continued operation until the scheduled February refuel outage.

1.0 Introduction

1.1 Purpose

The purpose of this safety assessment is to respond to Generic Letter (GL) 94-03, which requested a safety analysis supporting continued operation of Nine Mile Point 1 until the scheduled February 1995 refuel outage.

1.2 Scope

This Nine Mile Point 1 specific shroud cracking safety assessment uses the BWR Shroud Cracking Generic Safety Assessment[1] submitted to the NRC through the BWR owners group.

As requested in GL 94-03, this safety assessment includes details of the conditions that influence the probability of occurrence of cracking and the rate of crack growth at Nine Mile Point 1. Based on this information, the likelihood of shroud cracking in excess of the required structural margins is assessed and the uncertainty in the extent of cracking is reviewed and if appropriate, corrective actions identified. The safety assessment includes a probabilistic safety assessment to define the Nine Mile Point 1 overall risk assuming shroud weld failure. In addition, the shroud response to design basis loads assuming 360 degree through-wall cracking and the ability of the plant safety features to perform their design basis functions are reviewed. This safety assessment provides the basis for continued operation of Nine Mile Point 1 considering the uncertainty in the extent of cracking of the shroud welds identified in Figure 1-1.

1.3 Shroud Function and Weld Designations

The shroud is a stainless steel cylindrical assembly that provides a partition between the core region and the downcomer annulus, to separate the upward flow of coolant through the core from the downward recirculation flow. The shroud also provides, in conjunction with other components, a coolable core geometry. Nine Mile Point 1 relies on core spray cooling for the recirculation line LOCA and does not require the shroud to maintain a floodable geometry following a postulated recirculation line break (i.e. bottom entry recirculation lines preclude a floodable region). The shroud is not a primary pressure boundary component.

The following are the Nine Mile Point 1 shroud weld designations, also shown in Figures 1-1, 1-2,1-3 and 1-4.

H1,H2:	Upper welds, with H1 above and H2 below the emergency core cooling system (ECCS) injection.
H3:	Upper weld located below the bottom of the top guide support ring
H4,H5:	Mid-plane welds located above the core plate
H6A,H6B:	The welds located just above and below the core plate
H7:	Lower shroud to shroud support ring weld
H8:	Inconel 182 weld between 304 SS shroud support ring to inconel shroud support cone
H9:	Shroud Support Cone to Vessel Weld

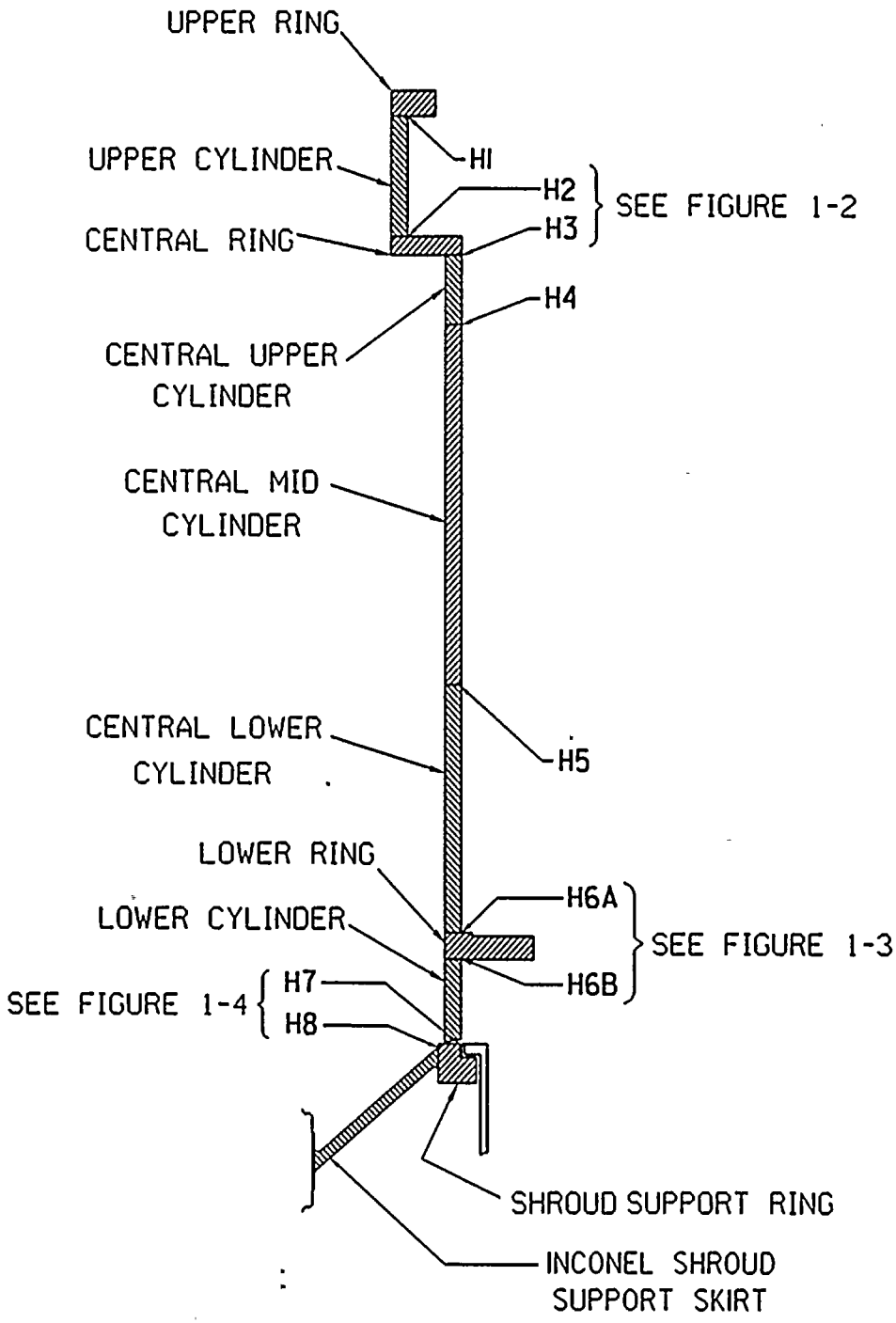


FIGURE 1-1 NINE MILE POINT 1 SHROUD WELDS

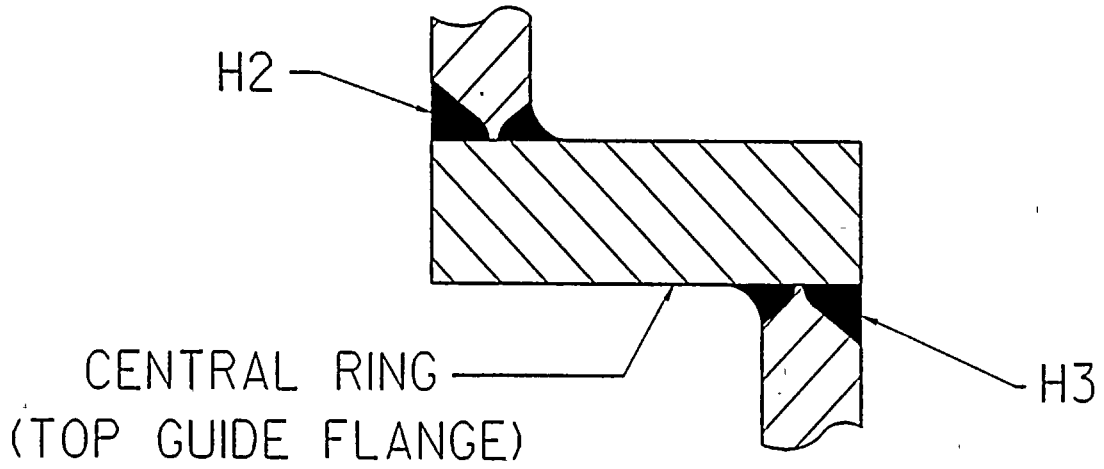


FIGURE I-2 DETAIL TOP GUIDE FLANGE

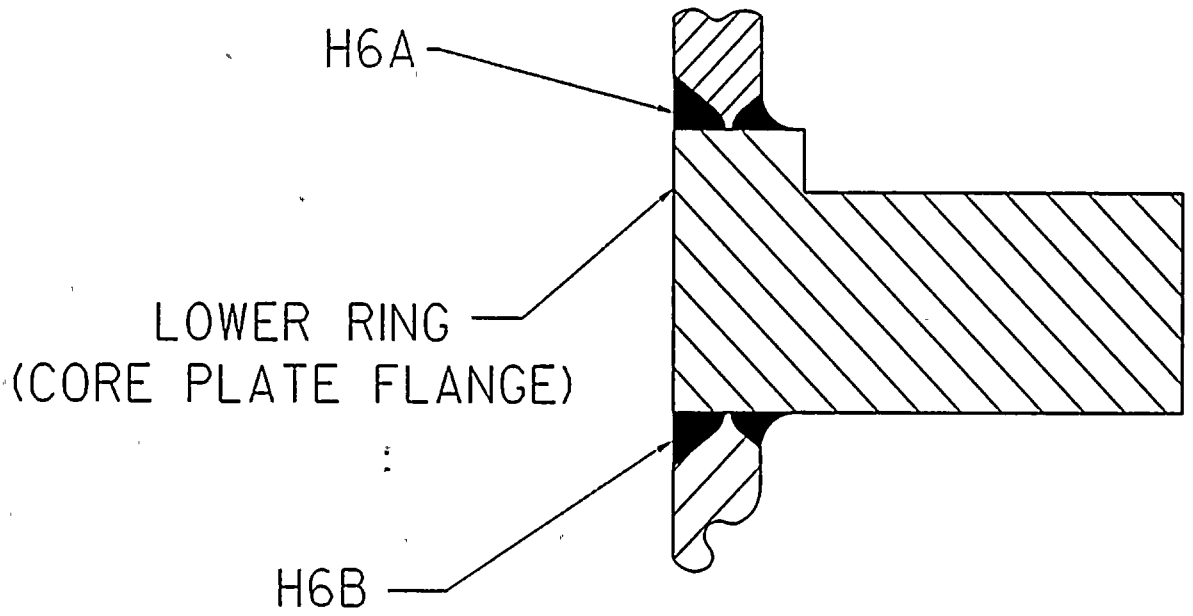


FIGURE I-3 DETAIL CORE PLATE FLANGE

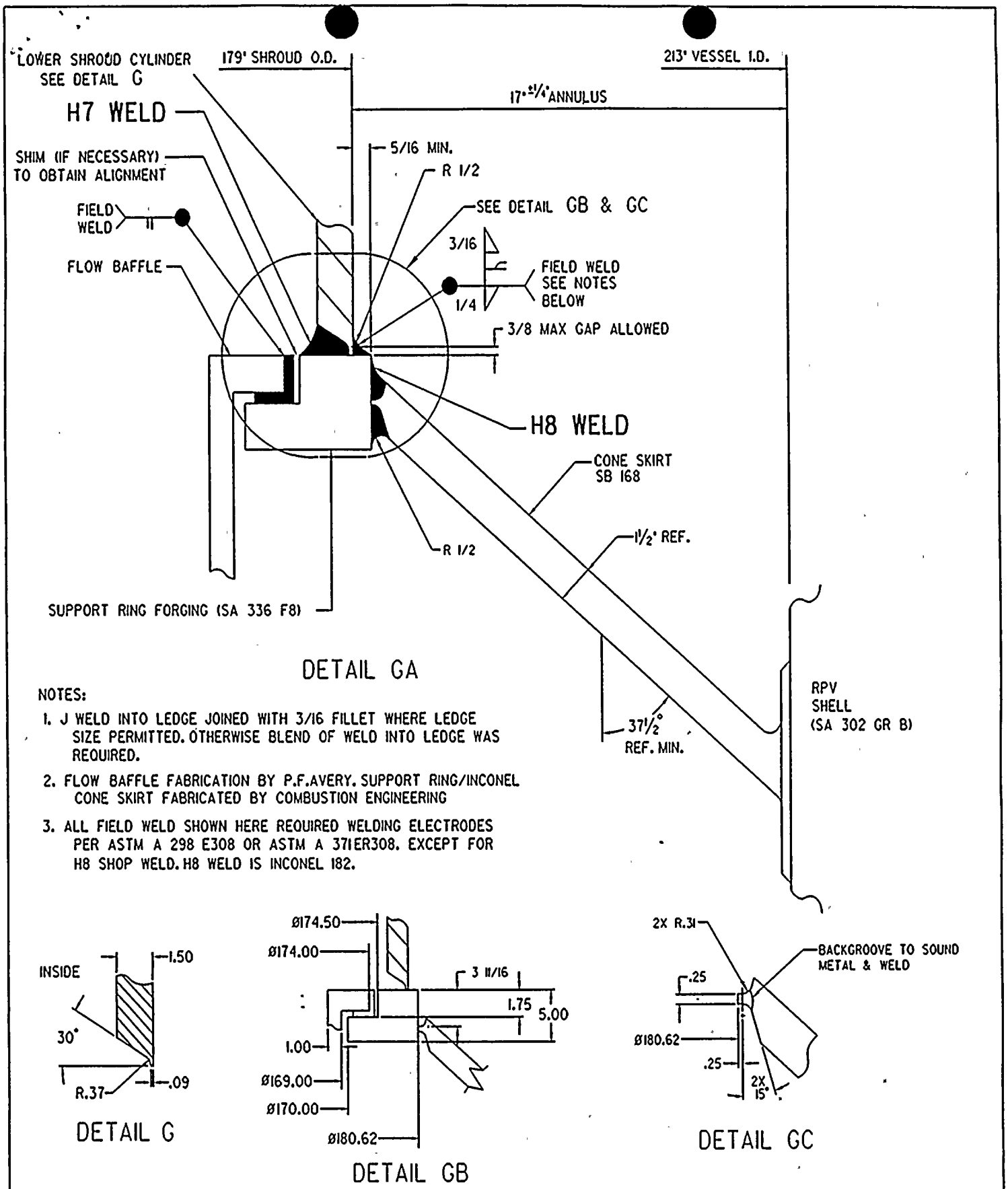


FIGURE 1-4 DETAIL SHROUD TO SHROUD SUPPORT JOINT

2.0 Susceptibility Assessment

2.1 Overview

The BWR Shroud Cracking Generic Safety Assessment[1] provided a discussion of the factors which contribute to the susceptibility of a shroud to stress corrosion cracking (IGSCC). The susceptibility criteria applied in reference 1, SIL 0572, R1[2] and the BWR Owners Group (BWROG) BWR Core Shroud Evaluation[3] are water chemistry, material carbon content, fabrication history, neutron fluence and hot operating time. The generic assessment recognized that factors such as degree of cold work and weld residual stress are significant factors affecting susceptibility; however, since quantitative information was not available these factors were not included. The Nine Mile Point 1 plant specific review of susceptibility discusses the above factors including residual stress, and available visual inspection information.

2.2 Nine Mile Point 1 Shroud Inspections

The generic assessment[1] discussed the inspection recommendations of SIL 0572, R1[2] and the BWROG BWR Core Shroud Evaluation[3] and provided both a qualitative summary of the inspection results and a quantitative summary of degree of cracking versus initial five cycle mean conductivity and cracking versus number of on-line years (hot operating time). This information, coupled with the susceptibility grouping factors, was used to establish the potential shroud welds for 360 degree cracking. Nine Mile Point 1 was classified in the generic assessment in the last grouping, 304 SS shrouds with welded plate rings and highest conductivity. The likelihood of 360 degree cracking for this grouping was considered fairly high, however, cracking greater than 90% through-wall was considered unlikely in the short term.

While Nine Mile Point 1 has not completed inspections in accordance with SIL 0572, R1, visual inspections of welds H7 and H8 have been performed as required by ASME Code Section XI visual inspection examination category B-N-1, Item B13.40, "Core Support Structure." These inspections satisfy the requirements of SIL 0572, R1 with the exception that no prior cleaning of the welds was performed. In addition, visuals of shroud welds H1,H2,H3 and H4 do exist as a result of access studies performed in 1989. The lighting provided was to the level needed for verification of access and did not meet the requirements of SIL 0572 R1; however, the welds were clearly visible. The camera resolution level was as needed to gage accessibility. No prior cleaning was performed. The shroud visual inspections are summarized in Table 2-4.

Niagara Mohawk is actively participating in the efforts of the BWRVIP inspection subcommittee to develop standardized visual inspection criteria. These standardized inspection criteria will include qualification of examination personnel through shroud specific experience and on-the-job training. In this regard, the inspection tapes summarized in Table 2-4 have been reviewed by an NMPC Level III qualified examiner concurrently with a GE level III qualified examiner, who participated in the examination activities at both Quad

Cities and Dresden and who is qualified to the enhanced level proposed by the new standards. Both examiners concluded that there was no evidence of cracking at the locations inspected.

2.3 Basis For Ranking Nine Mile Point 1 Shroud Welds

A Nine Mile Point 1 specific review of the fabrication history, water chemistry, material carbon content, neutron fluence, and on line years has been completed and is provided below.

2.3.1 Fabrication History

The generic assessment[1] identifies the shroud weld locations exhibiting the greatest extent of circumferential cracking as the ring to shell welds, i.e. H1,H2,H3, and H6A,H6B. The highest susceptibility was linked to the fabrication of rings cut from rolled plate and welded into a ring configuration, followed by machining to size. The Nine Mile Point 1 shroud rings, with the exception of the shroud support ring, are welded plate rings (see Figures 1-1 and 1-2). Table 2-1 has the details of the shroud materials. The fabrication records indicate that the Nine Mile Point 1 shrouds and the Oyster Creek shrouds were fabricated by P.F. Avery during the same time period. A specific breakdown between the two shrouds of which specific items went into each shroud could not be found, so all the heat numbers related to each part number are identified in Table 2-1. All shop P.F. Avery welds were submerged arc welds using ASTM A-371 Type ER-308 filler metal with 5% minimum ferrite content and a maximum interpass temperature of 350 degrees F.

The shroud support ring is a forged 304 SS ring, however, for Nine Mile Point 1 the H7 and H8 welds to the forged ring have a plant specific susceptibility because the shroud support ring forging was sensitized during the initial vessel heat treatment. The H7 weld is a field weld which was not stress relieved and therefore is considered to be suceptable to IGSCC cracking due to higher weld stresses and sensitized base material. The susceptibility to crack initiation and crack growth rate for the H7 weld is considered similar to that used for the welded plate ring evaluations (H1 through H6). The H8 location susceptibility is discussed in detail in Appendix A. The Appendix A analysis concludes that IGSCC through the heat affected zone (HAZ) of the H8 weld is extremely unlikely because the residual stresses were relieved for this weld during the initial vessel heat treatment, and because the H8 weld is generally compressive during operation. In addition, the Appendix A analysis predicts that IGSCC cracking at H8 would initiate at the OD surface due to the highest tensile stresses being located on the OD. The ASME XI inspection of the OD of H8 revealed no indications.

The inconel shroud support cone to vessel weld (H9) is not considered a shroud weld in this assessment. This is an inconel weld to inconel cone which is not creviced and was stress relieved during the vessel post weld heat treatment. This places the weld in a much more IGSCC resistant category which allows this weld to be eliminated from further discussion in this report.

2.3.2 Water Chemistry

The generic assessment[1] identifies the mean conductivity for the first five cycles as a factor for susceptibility grouping. Figure 2-1 and Table 2-2 provide the Nine Mile Point 1 specific cycle mean conductivities. This conductivity places Nine Mile Point 1 in the susceptible category for which IGSCC cracking in welded plate ring welds is likely to occur.

2.3.3 Material Carbon Content

The generic assessment[1] identifies the carbon content as a factor in susceptibility grouping, with 304 shrouds being more susceptible than 304L. The Nine Mile Point 1 shroud material is 304, with the specific material carbon content identified in Table 2-1.

2.3.4 Neutron Fluence

The generic assessment[1] did not select fluence as a primary contributor to extensive cracking. However, a fluence effect on cracking susceptibility (IASCC at f>3-5 E 20 nvt) or a synergistic interaction of fluence in already sensitized material (IGSCC at f>1E19 nvt) is expected and was verified at Brunswick-1 and KKM. The Nine Mile Point 1 specific fluence at each weld location is provided in Table 2-3.

2.3.5 On-Line Years

Consistent with the generic assessment[1], on-line years was used to estimate hot operating time. Nine Mile Point 1 on-line years is 14.4 years. The generic assessment did not use hot operating time to group the plants, however, the inspection data to date indicates that cracking in excess of 180 degrees was unlikely until a plant accumulated 10 on-line years.

2.4 Estimated IGSCC Susceptibility for Nine Mile Point 1

Consistent with the generic assessment[1], the likelihood of 360 degree cracking at some depth is fairly high. The likelihood of 360 degree cracking to depths approaching analysis allowables is considered unlikely and is discussed in Section 3.

TABLE 2-1 SHROUD MATERIAL TYPES

PART NUMBER	PART NAME	QUANTITY	MATERIAL			
			TYPE	HEAT NUMBER	CARBON CONTENT - %	COMMENTS ON MATERIAL / PROCESS
1	UPPER RING	2 PIECES	A240 TYPE 304	65444-1	.064	PLATE
2	UPPER CYLINDER	2 PIECES	A240 TYPE 304	65235-1A	.042	
				65235-1B	.042	
3	CENTRAL RING	2 PIECES	A240 TYPE 304	65294-1	.056	
4	CENTRAL UPPER CYLINDER	2 PIECES	A240 TYPE 304	65235-1	.042	
5	CENTRAL MID CYLINDER	2 PIECES	A240 TYPE 304	65291-1A	.052	
				840784-2	.053	
				65290-1	.047	
				65295-1	.062	
6	CENTRAL LOWER CYLINDER	2 PIECES	A240 TYPE 304	65290-1A	.047	
				840784-2B	.053	
				840784-2A	.053	
				65291-1	.052	
7	LOWER RING	2 PIECES	A240 TYPE 304	65444-1	.064	
8	LOWER CYLINDER	2 PIECES	A240 TYPE 304	65291-1	.052	
				840784-2A	.053	
				840784-2B	.053	
				65290-1A	.047	
9	SHROUD SUPPORT RING	1 PIECE	ASME SA-336 F8	G-23/245352	NOT KNOWN	FORGING

NINE MILE POINT UNIT 1 SHROUD DATA

NOTES: 1. HEAT NUMBER DATA SHOWN MAY BE APPLICABLE FOR NINE MILE POINT 1 AND/OR OYSTER CREEK.

Table 2-2
Nine Mile Point 1 Water Chemistry History

Cycle	Mean Value Conductivity µS/cm	Chloride ppb
1	0.432	30
2	0.525	46
3	0.591	58
4	0.445	44
5	0.291	33
6	0.225	27
7	0.181	26
8	0.133	25
9	0.087	18
10	0.082	1
11	0.084	1

Table 2-3
Shroud Weld Estimated Fluence

Estimated Fluences (n/cm^2, E>1Mev)

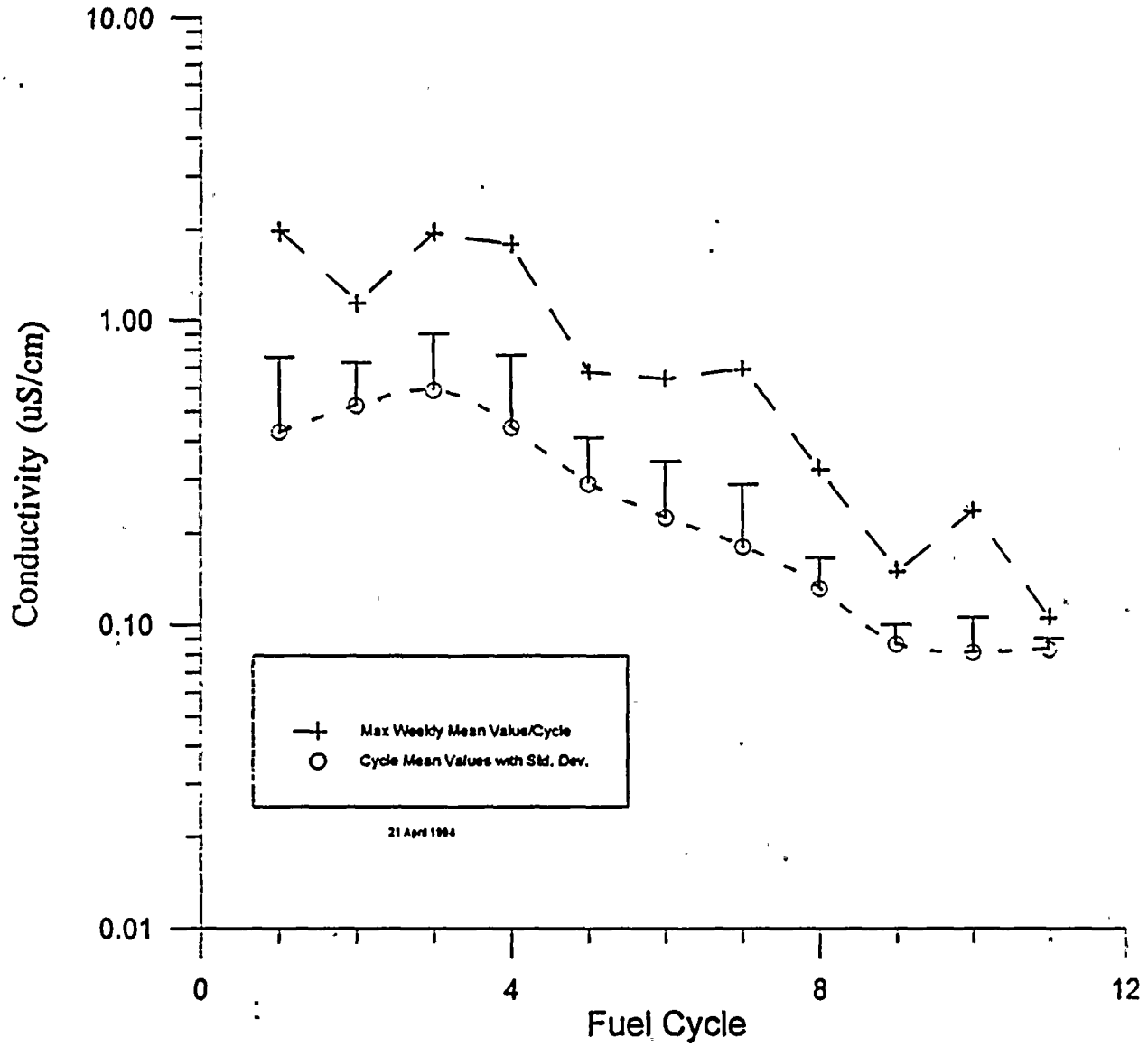
Weld	Estimated Flux $n/(sec-cm^2)$	Estimated Fluence** n/cm^2
H1	8.7E+09	4E+18
H2	4.6E+10	2E+19
H3	4.0E+11	2E+20
H4	7.8E+11	3.5E+20
H5	8.1E+11	3.6E+20
H6A	6.2E+07	3E+16
H6B	2.0E+07	9E+15
H7	<<1.0E+07	<<4E+15

**Based on projected 5197 EFPD at EOC Cycle 11 (EOC Cycle 11 February 1995)

Table 2-4 Shroud Visual Inspections

Weld Number	Date Exam'd	Length Exam'd	Approx % Examined	ID/OD Exam'd	Lighting Provided	Cleaned Yes/NO
H1	1989	63"	11%	OD	As needed	No
H2	1989	63"	11%	OD	As needed	No
H3	1989	29"	5%	OD	As Needed	No
H4	1989	85"	15%	OD	As Needed	No
H7	1986 1988 1993	568"	100%	OD	ASME XI SIL 572	No
H8	1986 1988 1993	568"	100%	OD	ASME XI SIL 572	No

Figure 2-1 Reactor Water Conductivity Mean Values



3.0 Structural Margin Assessment

The generic assessment[1] discussed the structural margins inherent in the shroud design and noted that 304 SS is a ductile material with high toughness properties even after accounting for the effects of neutron fluence, and that only a minimal remaining ligament (5%-10% of wall) is required to maintain structural margins under post accident loads when 360 degree cracking is present. The generic assessment applies an assumption that cracking is initiated after one fuel cycle and that crack growth can be estimated analytically using the PLEDGE Model. Inspection of the Nine Mile Point 1 hot operating time and mean cycle conductivity demonstrates that Nine Mile Point 1 is bounded by the generic assessment conclusion that finding a 360 degree through-wall crack with an average depth in excess of 90% during the Fall 1994 inspections is unlikely. However, because of the uncertainties associated with residual stress profiles and oxide wedging phenomenon, the generic assessment could not rule out cracking in excess of 90%. The uncertainty associated with the residual stresses is applicable to the H1 through H7 welds. The cracking uncertainty associated with H8 weld is addressed separately in Appendix A.

The next Nine Mile Point 1 refueling outage is currently scheduled for February 1995. A detailed inspection consistent with the BWRVIP inspection guidelines currently under development is scheduled and/or a pre-emptive repair would be implemented for all shroud horizontal welds. Considering the uncertainty in the extent of through-wall cracking prediction, the 1994 Fall outage inspection results of Oyster Creek (scheduled to commence at the end of September) will be used to determine the uncertainty associated with the potential for up to 360 degree cracking in excess of 90%. The Oyster Creek fall inspection is considered to be directly applicable to the structural margin assessment for Nine Mile Point 1 because the shroud fabrication is similar (see Section 2.3.1), and therefore, the uncertainty associated with residual stress profiles will be reduced. In addition, the Nine Mile Point 1 first 5 cycle mean conductivity is bounded by the Oyster Creek initial 5 cycle average, and the hot operating time is similar in magnitude. Therefore, the Oyster Creek inspection results are expected to allow Niagara Mohawk to assess the uncertainty associated with the structural margin integrity for the H1 through H7 welds.

4.0 Shroud Displacement Likelihood

As discussed in the generic assessment[1], several conditions must exist simultaneously in order for shroud displacement to occur. First, a 360 degree, > 90% deep crack must exist in the shroud. Then, a design basis guillotine main steam line break inside the flow limiters or a DEGB recirculation line break, or a design basis seismic event must occur to generate the loads assumed on the shroud. Note that operational transients, small and intermediate break LOCA, main steam line breaks outside the flow limiter and safe shutdown earthquake all have significantly reduced loads and minimal potential shroud displacement. In this section the Nine Mile Point 1 specific overall likelihood of the limiting design basis scenario is discussed using plant specific probabilities from the Individual Plant Examination (IPE).

4.1 Cracking Likelihood

Consistent with the generic assessment[1], it is considered unlikely that Nine Mile Point 1 has 360 degree cracking in excess of 90% depth. This is supported by the generic analysis and prior visual inspections. The uncertainty associated with this prediction is to be further clarified by review of the Fall Oyster Creek inspection (results expected in October 1994). In addition, 360 degree cracking in excess of 90% depth is not credible at the H8 inconel 182 weld between the shroud support ring and inconnel shroud support cone (see appendix A). Note that H8 is a limiting location in the generic assessment regarding the consequences of failure during a DEGB recirculation line LOCA.

4.2 LOCA Likelihood

A Nine Mile Point 1 specific probabilistic safety assessment (PSA) based on the Nine Mile Point 1 IPE was performed to evaluate the safety significance of continued operation prior to determining the status of welds related to the reactor shroud. The PSA evaluated the probability of accident scenarios resulting from potential failures of shroud welds. The events of primary concern are double ended guillotine breaks (DEGB) in main steam lines, DEGB in reactor recirculation lines, and earthquakes. Based on the probability of these events and the estimated conditional failure probability of the shroud welds, the estimated overall incremental core damage and large/early release frequency is less than 8.3E-8 per year. The detailed discussion of the PSA is included in Appendix B.

4.3 Inspection Timing

Consistent with the recommendations of the generic assessment[1], Nine Mile Point 1 plans to review the results of the Oyster Creek Fall inspection and other Fall inspections to assess the uncertainty associated with the evaluation that cracking is unlikely to be greater than 360 degrees and greater than 90% deep.

5.0 Integrated Shroud Assessment

The Nine Mile Point 1 susceptibility assessment has concluded that the likelihood of up to 360 degree cracking at some depth is fairly high at the H1 through H7 weld location. Cracking to any significant depth at the H8 weld location is considered extremely unlikely. The structural margin assessment established that cracking at the H1 through H7 locations in excess of 90% depth is unlikely. The uncertainty in the cracking prediction will be re-assessed based upon the results of the Oyster Creek Fall (October 1994) inspection.

The overall likelihood of the event scenario described, a LOCA or seismic event with a 360 degree, > 90% deep crack, is extremely unlikely. The assessment of the core damage frequency assuming shroud weld failure combined with the design basis seismic or recirculation line LOCA or MSLB LOCA is extremely low considering operation until February 1995, (4 E-8 per 6 months). This overall risk supports continued operation of Nine Mile Point 1 until the scheduled February 1995 refuel outage.

The consequences of 360 degree through-wall cracks applicable to Nine Mile Point 1 were reviewed as part of the generic assessment[1]. This assessment reviewed the shroud response to the structural loadings resulting from design basis events including, steam line break, recirculation line break and asymmetric loads associated with the recirculation line break. This assessment included a review of the ability of plant safety features to perform their functions considering the design basis accident loads with 360 degree through-wall cracking (e.g. control rod insertion, ECCS injection). Through the BWRVIP assessment subcommittee, analyses are under development which will provide more detailed shroud loads considering both the main steam line break LOCA and recirculation line LOCA. These analyses are intended to better define the asymmetric loads associated with the recirculation suction line LOCA and the amount of shroud lift following the main steam line break LOCA. These analyses are estimated to be completed in the October 1994 time frame. In the interim, additional information is available to supplement the generic assessment discussion of the recirculation line break and steam line break.

Main Steam Line Break Accident:

GPU Nuclear has completed preliminary assessments[4] using a RELAP 5 Oyster Creek model which confirms that the maximum potential lift is limited such that the top guide does not clear the fuel channels(e.g. less than 14 inches of lift) and control rod insertion is not expected to be impacted. Core spray lines are expected to be damaged by the possible displacement, however, the break is above TAF so ECCS injection inside the reactor vessel, at or above the core steaming rate, will assure short and long term cooling.

Recirculation Line Break Accident:

The generic assessment considered the shroud loads associated with the recirculation discharge line break as limiting and that

no vertical displacement is expected at any but the vertically unsupported H8 weld. The consequence of this failure was vertical displacement (downward) which would damage the core spray lines and result in impaired core spray cooling. However, additional study of the H8 weld configuration (see Appendix A) has determined that failure of the H8 weld in such a manner which would allow vertical downward displacement is not a credible failure. With the H8 weld integrity assured, the core spray system would perform its design basis function and control rod insertion is assured.

As discussed in the generic assessment, the lateral force on the shroud due to the blowdown asymmetric load is bounded by the restoring moment of the shroud weight and therefore, the recirculation line break analysis results are unchanged. As indicated above, additional analyses are in progress to address this issue generically through the BWRVIP assessment subcommittee.

Nine Mile Point 1 has also reviewed the guidance provided in the generic assessment regarding through-wall crack indication during normal operation. This generic information has been provided to operations and has been incorporated into the normal operating procedure for the Nuclear Steam Supply System (N1-OP-1). The procedure has been revised to alert operators of the expected plant response should a through-wall shroud crack develop. This training was provided to all operations crews prior to their resumption of shift duties.

6.0 Conclusions

The Nine Mile Point 1 shroud cracking safety assessment indicates that 360 degree, greater than 90% through-wall cracking is unlikely to occur during operation up to the scheduled February 1995 refuel outage. The uncertainty in this determination will be assessed based upon the results of the scheduled Fall 1994 Oyster Creek shroud inspection. Based on this conservative approach, the probability that the Nine Mile Point 1 core shroud does not satisfy the design basis structural integrity margins is considered extremely low. Even if the extreme condition of 360 degree, greater than 90% through-wall cracking coupled with a design basis accident is assumed, the safety assessment shows that control rod insertion is not expected to be impacted and the core spray system would provide adequate core cooling. This low probability, combined with the extremely low overall risk estimate, supports continued operation until the scheduled February 1995 refueling outage.

7.0 References

1) GENE-523-A107P-0794, Revision 1 "BWR Shroud Cracking Generic Safety Assessment", August 1994

2) SIL 0572 R1, "Core Shroud Cracks"

3) GE-NE-523-148-1193, "BWR Core Shroud Evaluation", April 1994

4) GPUN Calc #C1302-222-5450-W06, Rev 0, "Oyster Creek Shroud Dispacement Calculation"

5) NMPC Calc # S0-VESSEL-M025, rev 0, "Residual stress and remaining ligament calcualtion for the shroud ring H8 weld", Contains MPR calculation 085-252-01 referenced in Appendix A.

6) NMPC Technical Report SAS-94-005 "Probabilistic Risk Assessment of Potential Integranular Stress Corrosion Cracking of NMP1 Core Shroud", August 5, 1994

APPENDIX A

Analysis of Nine Mile Point 1 Weld H8 Shroud Support Ring to Inconel Shroud Support cone

MPR
ASSOCIATES INC.
ENGINEERS

August 16, 1994

ANALYSIS OF NINE MILE POINT UNIT 1
SHROUD SUPPORT RING TO INCONEL SUPPORT CONE WELD H-8

Weld H-8 attaches the Type 304 forged stainless steel shroud support ring to the Alloy 600 support cone as shown in Figure 1. The shroud support assembly, including the vessel shell, the cone and the ring, but not the shroud itself, was post weld heat treated (PWHT) at 1150°F during vessel fabrication. As a consequence, weld H-8 was stress relieved during PWHT, and peak tensile residual stresses were reduced by the heat treatment. In addition, the PWHT sensitized the stainless steel ring which increases concern for IGSCC on the ring side of weld H-8. While the weld metal and cone were also sensitized, these are Inconel materials which are more resistant to IGSCC than the stainless material, especially in the absence of crevices. Normal operating stresses in the support ring are expected to be compressive due to the difference in thermal expansion coefficients of the ring and the cone/vessel.

A finite element stress analysis was performed on the shroud support assembly to quantify normal operating stress levels and to determine the stress state during PWHT (MPR Calculation 085-252-01). From the latter analysis, the stress relief effects for presumed weld residual stresses can be estimated. The residual stresses that remain, when combined with operating stresses, determine the overall stress state of weld H-8 during normal operation.

The analysis confirmed that the hoop, radial and axial principal stress components are all generally compressive in the support ring during normal operation. The principal stress of concern is the radial one, since this stress component could lead to stress corrosion cracking whose orientation would cause weld H-8 to lose its vertical load-carrying capability.

Stress contour plots for the radial stress component during the original stress relief and under normal operating conditions are shown in Figures 2 and 3. The principal load during stress relief is the thermal load, during which time the shroud is not attached. During normal operation, the loads include thermal, pressure, deadweight and hydraulic uplift forces.

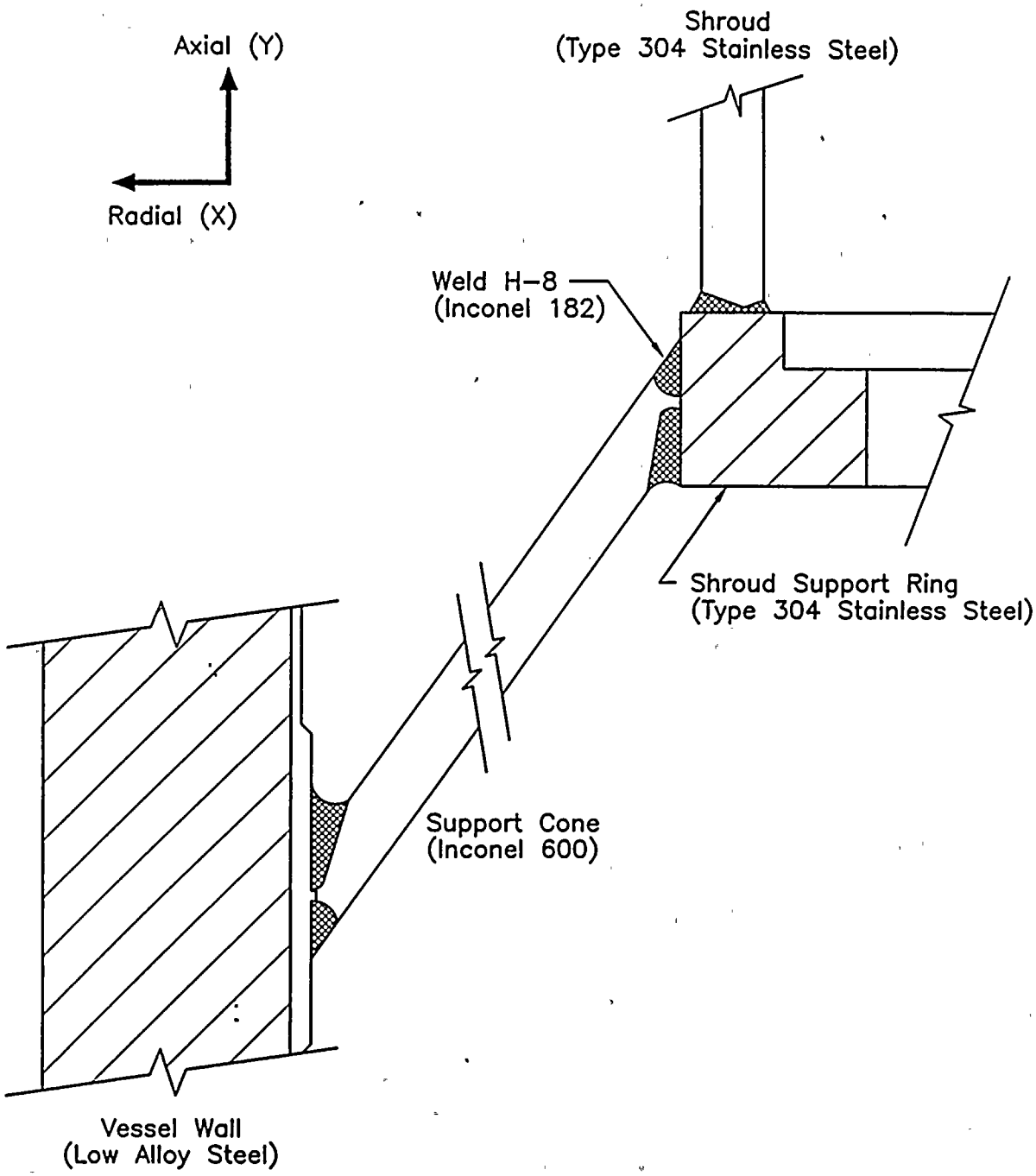
Inspection of Figure 3 reveals that the highest tensile radial stresses exist at the top of weld H-8 in the ring. This is considered to be the limiting location for possible crack initiation. It is presumed that local tensile weld residual stresses existed at this location after welding, and they were at the yield stress level. During stress relief (see Figure 2),

induced tensile thermal stresses, combined with the reduced yield strength at the 1150°F temperature, reduced the weld residual stresses to a low value, about 7 ksi. During normal operation, these stresses add to the normal operating stresses for a total stress of about 13 ksi at the top of weld H-8 in the ring.

While it is unlikely that cracking would initiate in a 13 ksi stress field, we note that the location of highest tensile stress is an area that is inspectable. Further, cracking in the heat affected zone of the support ring adjacent to weld H-8 would grow very slowly, if at all, because: (1) it would have to grow into an area of applied compressive stresses, and (2) weld residual stresses are also expected to become compressive in the ring at the center of weld H-8.

Finally, the limiting load for weld H-8 is the recirculation pipe break download of 2 million lbs. It is estimated that a ligament of only 1/4 inch is required in weld H-8 to support this load. Therefore, cracking in weld H-8 would have to be quite extensive, about ninety percent throughwall for 360 degrees, in order for the weld to fail under limiting accident conditions.

We conclude that, because of reduced weld tensile residual stresses in weld H-8, and the fact that only about 1/4 inch of weld is required to support the limiting download transient, weld H-8 is extremely unlikely to be in a condition that could fail.



MPR
F-C85-252-01
08/16/94 (131)

Figure 1. Shroud Support Components and Weld H-8

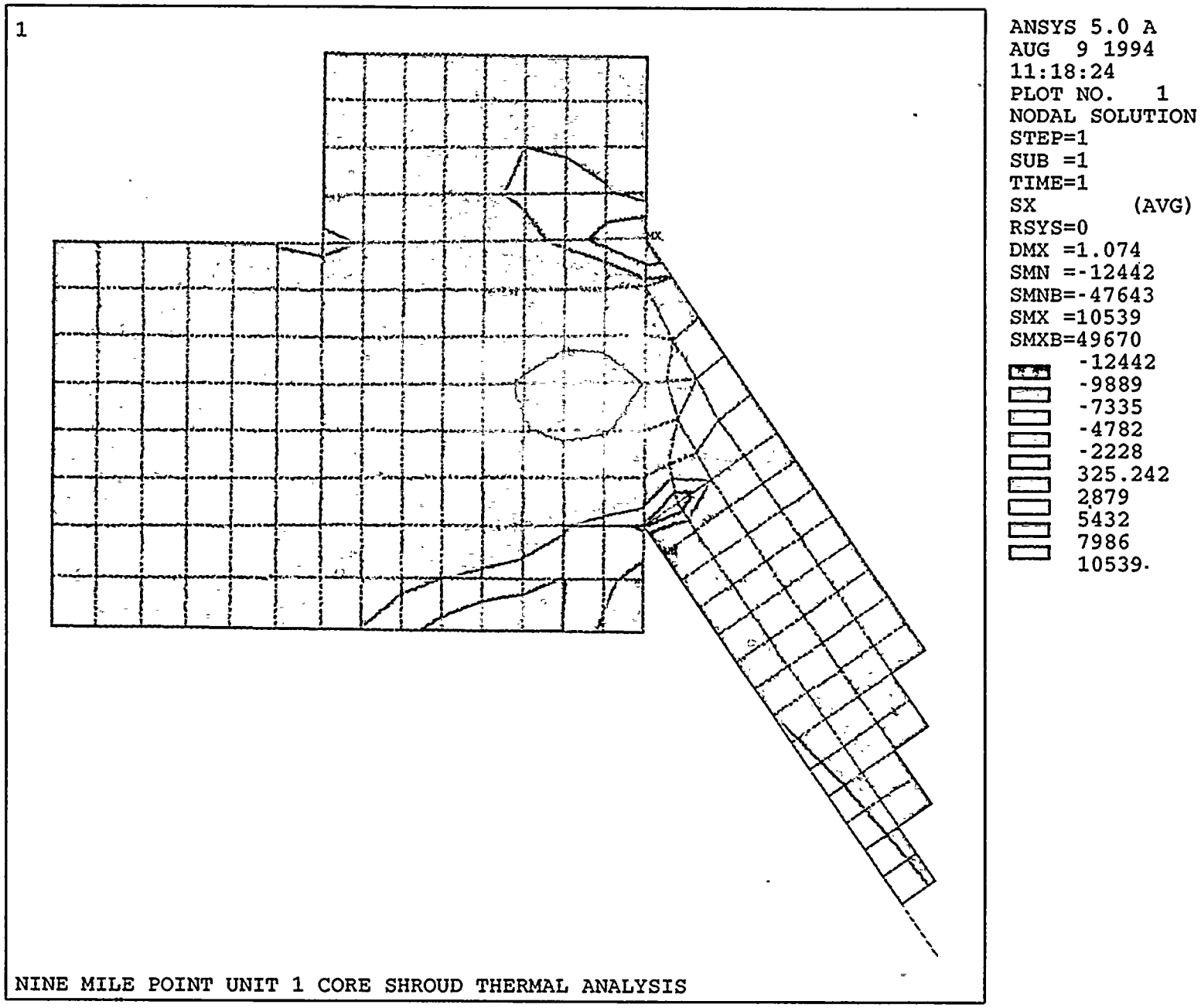


Figure 2. Radial Stress During PWHT

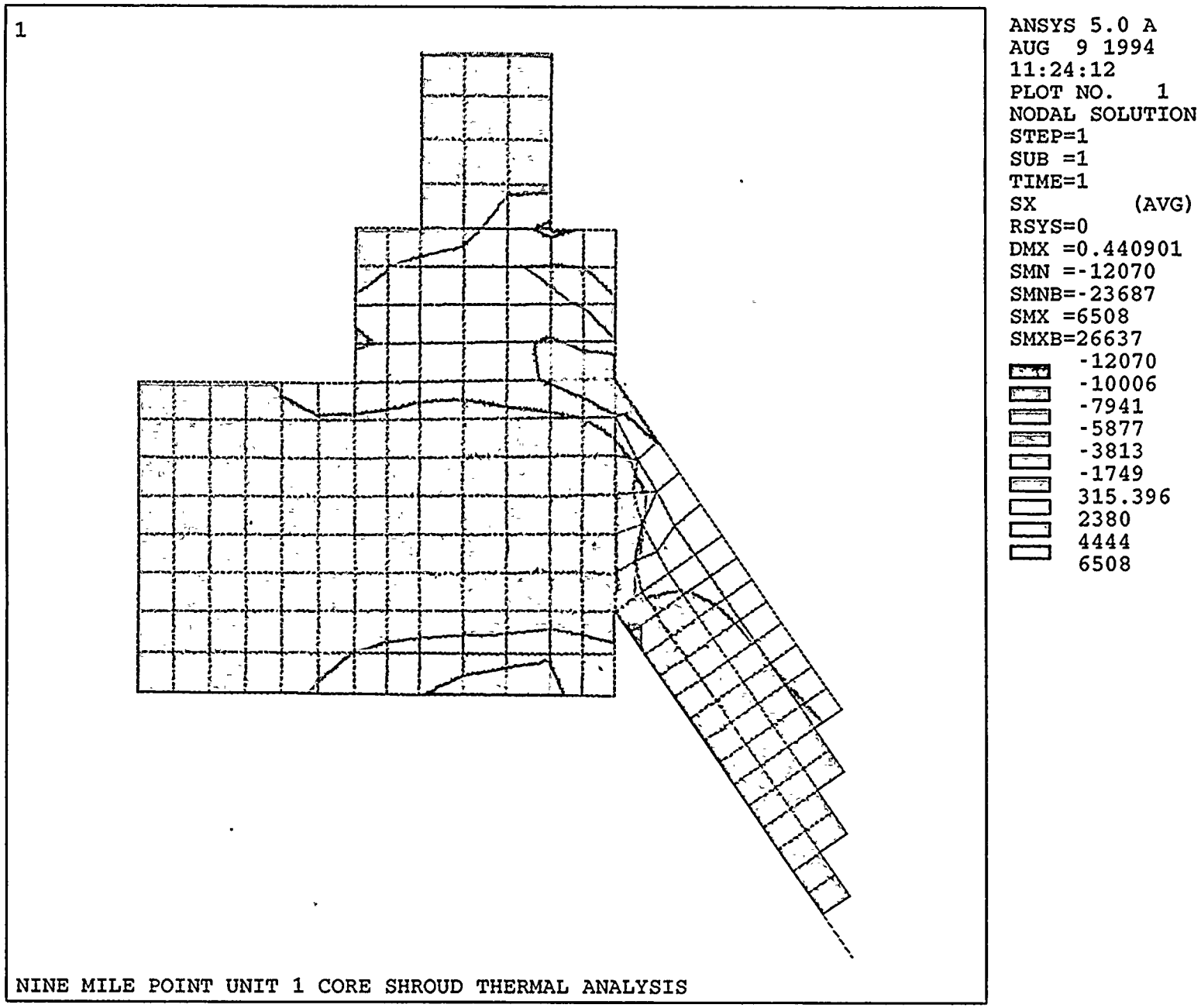


Figure 3. Radial Stress During Normal Operation

APPENDIX B

Analysis and Results from NMPC Technical Report SAS-94-005 "Probabilistic Risk Assessment of Potential Integranular Stress Corrosion Cracking of NMP1 Core Shroud", August 5, 1994

The cracking of vessel shroud welds can lead to failure of vessel internals support. Should vessel support fail, core integrity may be compromised and, of chief importance, insertion of control rods may not be possible if fuel bundles are no longer parallel to control rod movement. In addition, for NMP1, shroud movement may result in failure of core spray spargers during some events. Based on the engineering review and the information in References 1, 2, and 3, it is apparent that the principal scenarios of concern are main steam line breaks (MSLB), reactor recirculation line breaks (RRLB), and seismic events. Each of these events has the potential to induce loadings that could fail cracked welds and cause core movement. As such, this report calculates the probability of each of the events. These probability values should be useful in assessing the safety significance of the issue at NMP1.

The NMP1 IPE is a detailed evaluation of the probability and consequences of plant risk. Because it was completed prior to the elevated concern regarding shroud IGSCC, it does not explicitly include recent insights associated with this issue. In that regard, the probability of shroud IGSCC events can be calculated, similar to the events developed in the IPE, and presented as an incremental risk above-and-beyond that calculated in the IPE.

The best way to develop and describe the risk associated with the postulated IGSCC events is to treat each possible scenario separately and then sum the probability of each scenario to develop the total IGSCC event frequency. Therefore, the following section individually treats the MSLB, RRLB, and seismic events.

Main Steam Line Break (MSLB) Shroud IGSCC Risk

The MSLB event is defined by the rapid initiation of a 360° circumferential crack which results in a double ended guillotine break (DEGB) of piping and an immediate loss of coolant (LOCA) event. As such, makeup flow, from primarily the core spray system and feedwater, is required. This specific event was modeled in the NMP1 IPE as one of several contributors to the large LOCA (LLOCA) class of initiators. Other contributors to the large LOCA event frequency include: core spray system leakage/ruptures, multiple instrument penetration failures, SLC system piping leakage/rupture, and feedwater piping leakage/rupture. As discussed above, non-DEGB events pose little threat relative to the shroud issue and are not included in this risk assessment. Large LOCA events, including DEGB, were not a significant contributor to IPE calculated accident frequency.

The postulated MSLB initiating event frequency represents a certain fraction of the IPE LLOCA initiating event frequency since MSLB is one of several events that are considered to cause a LLOCA. However, the NMP1 IPE does not differentiate between specific LLOCA events because, for the purposes of severe accident modeling, the plant response is similar between individual events that fit the IPE LLOCA definition. As the NMP1 IPE LLOCA frequency is quantified as 7E-4 per year, for this analysis, the MSLB frequency can be reasonably considered less

than 7E-4 per year.

The BWROG[6] has performed research that is useful in characterizing the extent to which MSLB frequency is less than NMP1 LLOCA calculated frequency. The BWROG estimates that recirculation system piping, and it is inferred other large piping, has a rupture frequency of "several orders of magnitude lower than" 7.51E-6 per year.

It should be pointed out that there is considerable uncertainty as to the possibility of an instantaneous DEGB of pipes in a nuclear application. A widely considered theory suggests that the pipe would leak for some time before catastrophic rupture. As such, the leak would alert operators who would shut-down the plant before the DEGB occurred. In any event, no credit is taken for the leak-before-break argument and the BWROG value is assumed representative of the MSLB frequency under the DEGB failure mode.

Following a MSLB event, the integrity of the shroud welds will be challenged. IPEs have typically assumed that this passive failure mode is of low probability. However, due to the IGSCC issue, this assumption, at least temporarily, should be questioned. Per Pinelli[3], the possibility that weld failures will occur is "unlikely" even considering IGSCC. For this analysis, "unlikely" is reasonably translated to mean less than 1E-2 per event. As such, the probability of a MSLB and resultant shroud failure is estimated as less than 7.51E-8 per year.

According to the current IPE model, this event would be considered a core damage and large-early release event since any large LOCA with a failure to SCRAM was connected to core damage and large/early release endstates. When added to a core damage frequency (CDF) of 5.5E-6 per year and a large/Early release frequency (LERF) of 6.9E-7 per year, the MSLB-Shroud event probability, 7.51E-8 per year, is a relatively minor contributor.

In actuality, the impact is even less than the above calculation shows. The NMP1 IPE model does not link LLOCA events with a failure to SCRAM to the ATWS model but rather to a Class IV failure endstate. The IPE conservatively assumes that such scenarios result in core damage. This is done because the low probability of the sequences does not justify the level of effort required to incorporate the necessary modeling details.

In reality, even if control rods are not inserted, the reactor could possibly be shut down using SLC injection. This is especially appropriate to consider since, at a minimum, some rods may be at least partially inserted even with the fuel rods in disarray. From the IPE, SLC failure probability, including the associated operator actions to initiate SLC and prevent dilution (Top events SL, EP, and CH), is 1.6E-3 per event. Multiplying this by the above 7.51E-8 per year yields an event probability of 1.2E-10 per year. As such, if SLC could be credited then the risk impact would be far less. Note that this does not include any consideration of SLC equipment or operator failures that could relate to a relocated shroud. In any event, the above shows the nature of safety provided by the SLC system during the postulated MSLB/shroud event.

In addition, even if SLC failed, it is possible to keep the core covered and establish heat removal equal to or greater than that generated by the reactor. This could be accomplished by some combination of containment spray or containment vent operation in concert with some combination of successful injection. These actions would be directed via the current symptom based emergency operating procedures (EOP). It is not suggested that the success probability would be large, but rather, it is pointed out to show the conservatism in the above calculations.

Reactor Recirculation Line Break (RRLB) Shroud IGSCC Risk

The RRLB event initiates when a 360° circumferential crack develops in a reactor recirculation line and very quickly ruptures. As with the MSLB, this event results in an immediate loss of coolant (LOCA) event. The IPE treatment relative to large LOCA is the same as that discussed above. As such, all related discussion for MSLB is relevant here.

From above, a good estimate, for the purposes of this study is the BWROG 7.51E-6 per year value. Should a RRLB event occur, makeup is immediately required. However, since the break could be below the core, core spray is required because spray cooling is necessary to protect the core until containment flooding is completed. For NMP1, the core spray spargers are attached to the shroud. As such, following a RRLB, should the H8 shroud core support weld fail, the entire shroud could drop. This would result in failure of the core spray function; although core spray flow will still reach the vessel.

Due to the nature of the H8 shroud weld, engineering review has determined that it is very reliable; even considering the IGSCC issue. This review has classified the H8 weld failure as an extremely unlikely event. As such, a value of 1E-3 per event failure frequency has been conservatively assigned as the failure probability of the H8 shroud weld following a RRLB event. Multiplying 1E-3 per event by the above 7.51E-6 per year yields a RRLB/Shroud/Core spray failure event of 7.51E-9 per year.

In addition, the above is potentially conservative because although the core spray function is failed, its inventory, as injection rather than spray, can reach the vessel. Combined with feedwater, CRD, and SLC, the total inventory might be enough to prevent fuel damage prior to completion of containment flooding. This success path is not credited in this, or most likely any other analysis, but it is mentioned here as a possible success path that could be developed further.

Seismic Event Shroud IGSCC Risk

Per NRC[2], the safe shutdown earthquake (SSE) is expected to produce minimal movement of equipment related to the above issue. As such, it would require MSLB or RRLB events coincident with the SSE to cause potential accidents. Due to the low likelihood of each individual event, the coincident occurrence of the SSE and a large break is

considered very small. Even if the SSE and large break could cause a problem (assume they occurred on the same day rather than in the same minute or hour), a CDF estimate of less than 1E-10 per year would result.

Additionally, a beyond SSE earthquake could also occur such that the earthquake itself causes failure of a large line (MSLB or RRLB) and the shroud. Events of this magnitude are of very low probability (i.e. <1E-7 per year). Also, earthquakes of this magnitude would likely fail a significant portion of other plant equipment such that conditional failure probability of the plant as a whole would be large regardless of the status of the shroud welds. As such, the incremental risk caused by potentially cracked shroud welds is judged insignificant.

Summing the above conservatively calculated scenario frequencies results in a total incremental CDF and LERF frequency of 8.27E-8 per year. Considering that only six months remain until the next refueling outage the incremental risk is half that above, or 4.14E-8 per 6 months. Also, it has been demonstrated that the above calculation is conservative. Conservative or not, the above incremental risk is very small.

REFERENCES

(1) Zimmerman, R.P. "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," Generic Letter 94-03, USNRC, July 25, 1994.

(2) Stang, J.F. (USNRC) Letter to Farrar, D.L. (ComEd), "Resolution of Core Shroud Cracking at Dresden, Unit 3 and Quad Cities, Unit 1," 7/21/94.

(3) Pinelli, R.A. Letter to BWR Owner's Group Executives, "BWROG Response to NRC Request for Shroud Information," GE-NE-523-AI07P-0794, July 13, 1994. *(GE Proprietary.)*

(4) Nine Mile Point Unit One Final Safety Analysis Report (Updated).

(5) Kirchner, R.F., et. al., "Nine Mile Point Nuclear Station - Unit 1 Individual Plant Examination (IPE)" Niagara Mohawk Power Corporation, SAS-TR-93-001, July 1993.

(6) England, L.A. Letter to USNRC (Serkiz, A.W.), "Response to NRC Request for Information on Pipe Break Frequencies," BWR Owner's Group, BWROG-93149, December 8, 1993.

ATTACHMENT B

NINE MILE POINT UNIT 1
DOCKET NO. 50-220
LICENSE NO. DPR-63

GENERIC LETTER 94-03

HISTORY OF CORE SHROUD INSPECTIONS

Niagara Mohawk has not performed any shroud inspections which meet all of the General Electric (GE) SIL 572, Revision 1 recommendations for lighting level, camera resolution and weld pre-cleaning. However, visual (camera) inspections of vessel internal components, which directly or indirectly included film footage of the shroud plates and weldments, have previously been performed. These inspection tapes have been reviewed jointly by a Niagara Mohawk Level III qualified examiner and a General Electric Level III qualified examiner, who participated in the examination activities at both Quad Cities and Dresden and who is qualified to the enhanced level proposed by the new standards. The results of Niagara Mohawk's review of past invessel visual inspection tapes is provided below and summarized in the accompanying table.

Core shroud welds H1, H2, H3 and H4 were filmed during a 1989 access study conducted to verify clearances for the reactor pressure vessel beltline inspection tool. The lighting provided was to the level needed for verification of access, and though not specifically deployed to illuminate the welds, the welds are clearly visible. The camera resolution level was as needed to gauge accessibility. No prior cleaning of the welds was performed. The review of these tapes by the Level III qualified examiners concluded that there was no evidence of gross cracking at the locations inspected.

Invessel inspections of the shroud supporting ring to the shroud support skirt (ISI component RV15I) were conducted in 1986, 1988 and 1993 in accordance with ASME Section XI. The areas inspected encompass core shroud welds H7 and H8. The lighting level and camera resolution required by ASME Section XI meet the requirements of GE SIL 572, Revision 1. No prior cleaning of the welds was performed. The inspections of ISI component RV15I in 1986, 1988 and 1993 did not reveal any reportable indications.

NMP1 CORE SHROUD VISUAL (CAMERA) EXAMINATIONS

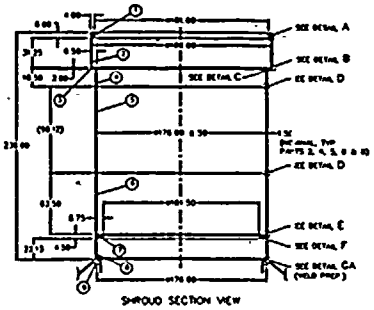
Weld Number	Date Exam'd	Length Exam'd	Approx % Examined	ID/OD Exam'd	Lighting Provided	Cleaning Yes/No
H1	1989	63"	11%	OD	As needed	No
H2	1989	63"	11%	OD	As needed	No
H3	1989	29"	5%	OD	As needed	No
H4	1989	85"	15%	OD	As needed	No
H7	1986 1988 1993	568"	100%	OD	ASME XI/ SIL 572	No
H8	1986 1988 1993	568"	100%	OD	ASME XI/ SIL 572	No

ATTACHMENT C

NINE MILE POINT UNIT 1
DOCKET NO. 50-220
LICENSE NO. DPR-63

GENERIC LETTER 94-03

DRAWINGS OF CORE SHROUD CONFIGURATION



SHROUD SECTION VIEW

NOTES:

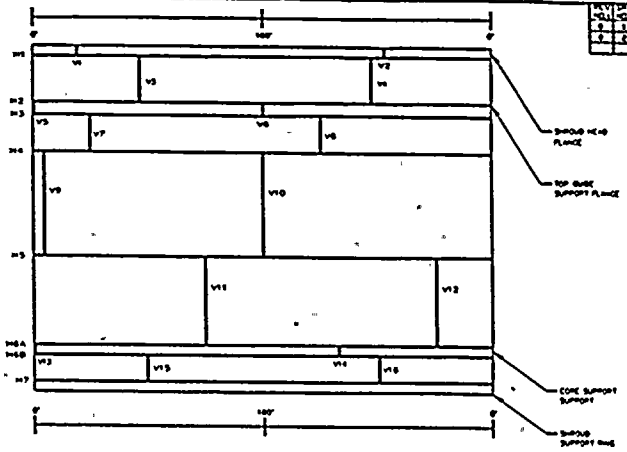
1. CORE SPRAY SPARGER ASSEMBLY, INTERNAL AND EXTERNAL BRACKETS/LUGS ARE NOT SHOWN FOR CLARITY.
2. ALL DIMENSIONS ARE REFERENCE.

HORIZONTAL WELD DETAILS	
HORIZONTAL WELD NO.	WELD DETAIL NO.
H1	DETAIL A
H2	DETAIL B
H3	DETAIL C
H4	DETAIL D
H5	DETAIL D
H6A	DETAIL E
H6B	DETAIL F
H7	DETAIL GA
H8	DETAIL GA

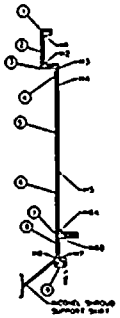
VERTICAL WELD DETAILS	
VERTICAL WELD NO.	WELD DETAIL NO.
V1, V2	DETAIL H
V3, V4	DETAIL J
V5, V6	DETAIL K
V7, V8	DETAIL L
V9, V10	DETAIL L
V11, V12	DETAIL L
V13, V14	DETAIL M
V15, V16	DETAIL M

NOTES:

1. WELD PREPARATIONS FOR THE RINGS ARE VIEWED FROM THE SIDE OF THE SHROUD FOR PARTS 2 & 7, AND FROM THE TOP FOR PART 1.
2. WELD PREPARATIONS FOR THE CYLINDERS ARE VIEWED FROM THE TOP OF THE THE SHROUD FOR PARTS 3, 4, 5, & 6.
3. ATTACHING LOCATIONS OF LONGITUDINAL WELDS ARE UNKNOWN



SHROUD SURFACE DEVELOPMENT

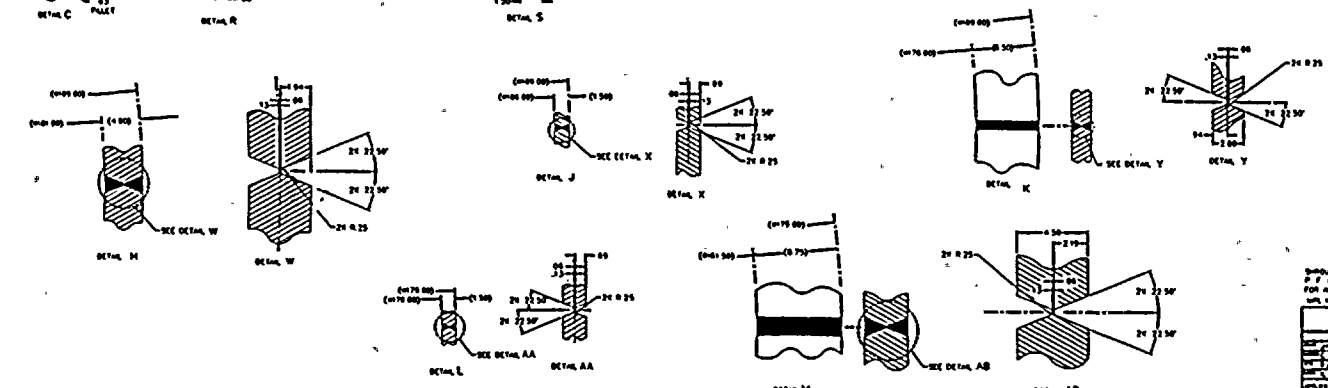


PART NUMBER	PART NAME	QUANTITY	MATERIAL				
			TYPE	HEAT NUMBER	CARBON CONTENT %	COMMENTS ON MATERIAL / PROCESS	
1	UPPER RING	2 PIECES	A240 TYPE 304	85444-1	.054	PLATE	
2	UPPER CYLINDER	2 PIECES	A240 TYPE 304	85235-14 85235-18	.042 .041		
3	CENTRAL RING	2 PIECES	A240 TYPE 304	85294-1	.056		
4	CENTRAL UPPER CYLINDER	2 PIECES	A240 TYPE 304	85235-1	.042		
5	CENTRAL MID CYLINDER	3 PIECES	A240 TYPE 304	85294-14 84478-1 85294-1 85294-2	.047 .051 .047 .047		
6	CENTRAL LOWER CYLINDER	3 PIECES	A240 TYPE 304	85290-16 84478-10 84478-11 84478-13 85294-1	.047 .051 .051 .051 .047		
7	LOWER RING	2 PIECES	A240 TYPE 304	85444-1	.054		
8	LOWER CYLINDER	2 PIECES	A240 TYPE 304	85294-1 84478-11 84478-13 84478-19 85294-14	.047 .051 .051 .051 .047		
9	SHROUD SUPPORT RING	1 PIECE	ASME SA-336 F6	C-23/P-5352	NOT KNOWN	FORGING	

NINE MILE POINT 1 SHROUD DATA

NOTES:

1. HEAT NUMBER DATA SHOWN MAY BE APPLICABLE FOR NINE MILE POINT 2 AND/OR BWR/6 CHECK



SHROUD FABRICATOR
P. F. SHEET
FOR ADDITIONAL FABRICATION DATA SEE REPORT GENE 77-20-0404
MFG NO.: 0004-0013 (0013)

NUCLEAR ENERGY

NINE MILE POINT 1
SHROUD DATA

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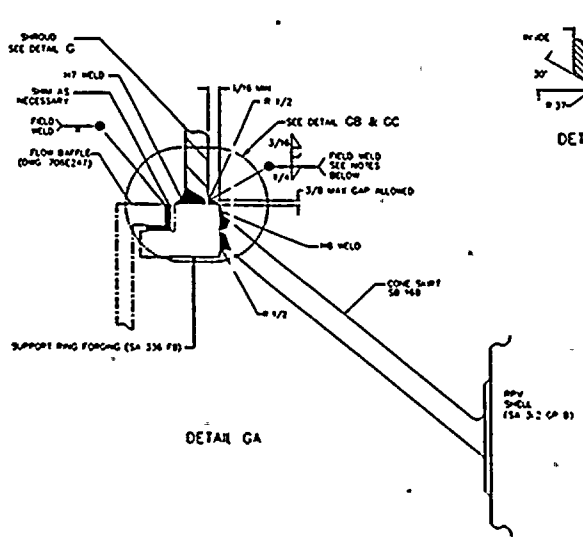
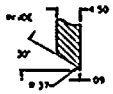
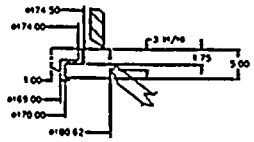


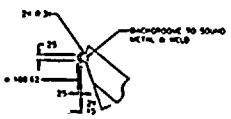
DETAIL CA



DETAIL C



DETAIL CB

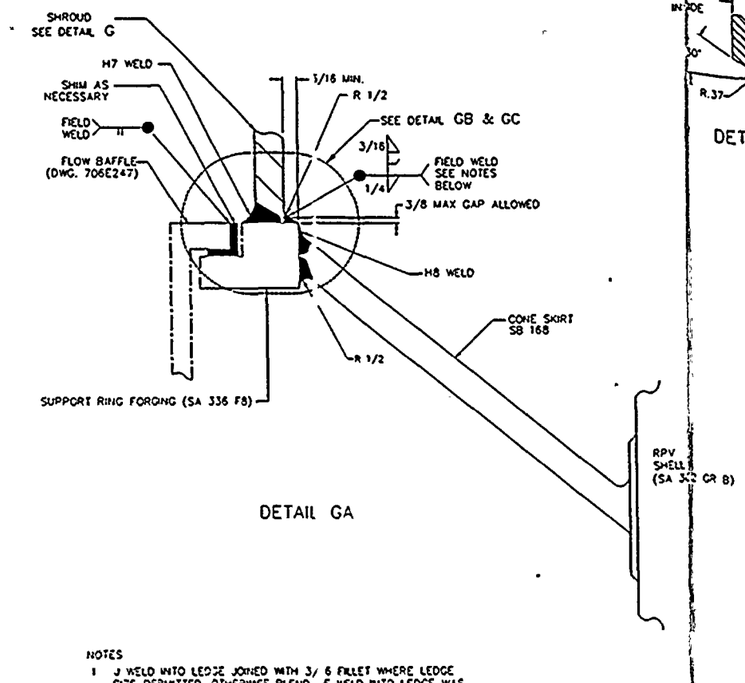


DETAIL CC

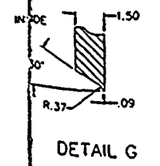
NOTES

1. J WELD INTO LEDGE JOINED WITH 3/8 FILLET WHERE LEDGE SIZE PERMITTED SHEPHERD BLEND F WELD INTO LEDGE WAS REQUIRED
2. FLOW BAFFLE FABRICATION BY PEJAERY SUPPORT RING/INCONEL CONE SHIFT FABRICATED BY COMBUSTION ENGINEERING
3. ALL FIELD WELD SHOWN HERE REQUIRED WELDING ELECTRODES PER ASTM A 293 E308 OR ASTM A 371 E308 (EXCEPT FOR H8 WELD)

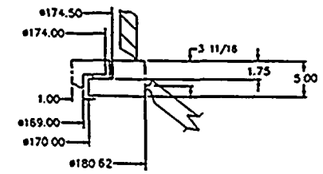
105E1413A



DETAIL GA



DETAIL G



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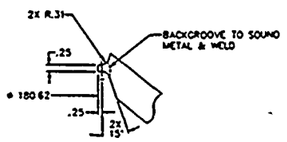

DETAIL GC

NOTES

1. J WELD INTO LEDGE JOINED WITH 3/ 6 FILLET WHERE LEDGE SIZE PERMITTED. OTHERWISE BLEND F WELD INTO LEDGE WAS REQUIRED
2. FLOW BAFFLE FABRICATION BY P.F.J VERY SUPPORT RING/INCONEL CONE SKIRT FABRICATED BY COMBU TION ENGINEERING
3. ALL FIELD WELD SHOWN HERE REQU IRED WELDING ELECTRODES PER ASTM A 293 E308 OR ASTM A 371 ER308. (EXCEPT FOR H8 WELD)

9409010263-01

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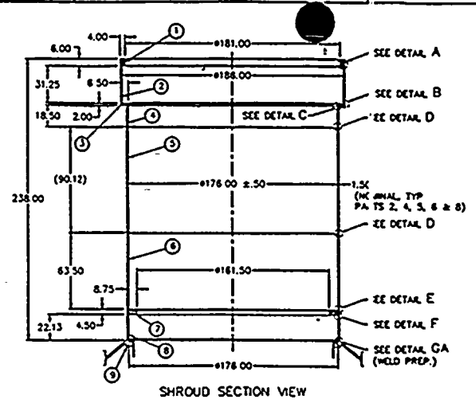
CL. EPOLITE	29 07 94	JUL 29 1994	SC 2

83706020



SHROUD SECTION VIEW

NOTES:

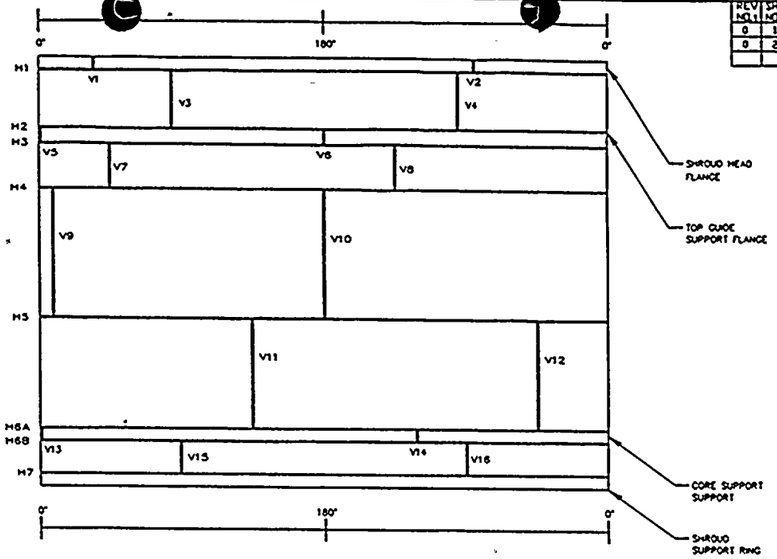
1. CORE SPRAY SPARGER ASSEMBLY, INTERNAL AND EXTERNAL BRACKETS/PADS ARE NOT SHOWN FOR CLARITY.
2. ALL DIMENSIONS ARE REFERENCE.

HORIZONTAL WELD DETAILS	
HORIZONTAL WELD NO.	WELD DETAIL NO.
H1	DETAIL A
H2	DETAIL B
H3	DETAIL C
H4	DETAIL D
H5	DETAIL D
H6A	DETAIL E
H6B	DETAIL F
H7	DETAIL GA
H8	DETAIL GA

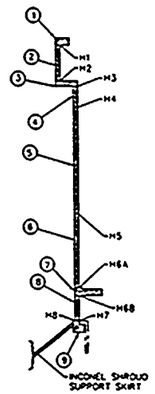
VERTICAL WELD DETAILS	
VERTICAL WELD NO.	WELD DETAIL NO.
V1, V2	DETAIL H
V3, V4	DETAIL J
V5, V6	DETAIL K
V7, V8	DETAIL L
V9, V10	DETAIL L
V11, V12	DETAIL L
V13, V14	DETAIL M
V15, V16	DETAIL L

NOTES:

1. WELD PREPARATIONS FOR THE RINGS ARE VIEWED FROM THE SIDE OF THE SHROUD FOR PARTS 3 & 7, AND FROM THE TOP FOR PART 1.
2. WELD PREPARATIONS FOR THE CYLINDERS ARE VIEWED FROM THE TOP OF THE THE SHROUD FOR PARTS 2, 4, 5, 6 & 8.
3. AZIMUTHAL LOCATIONS OF LONGITUDINAL WELDS ARE UNKNOWN.



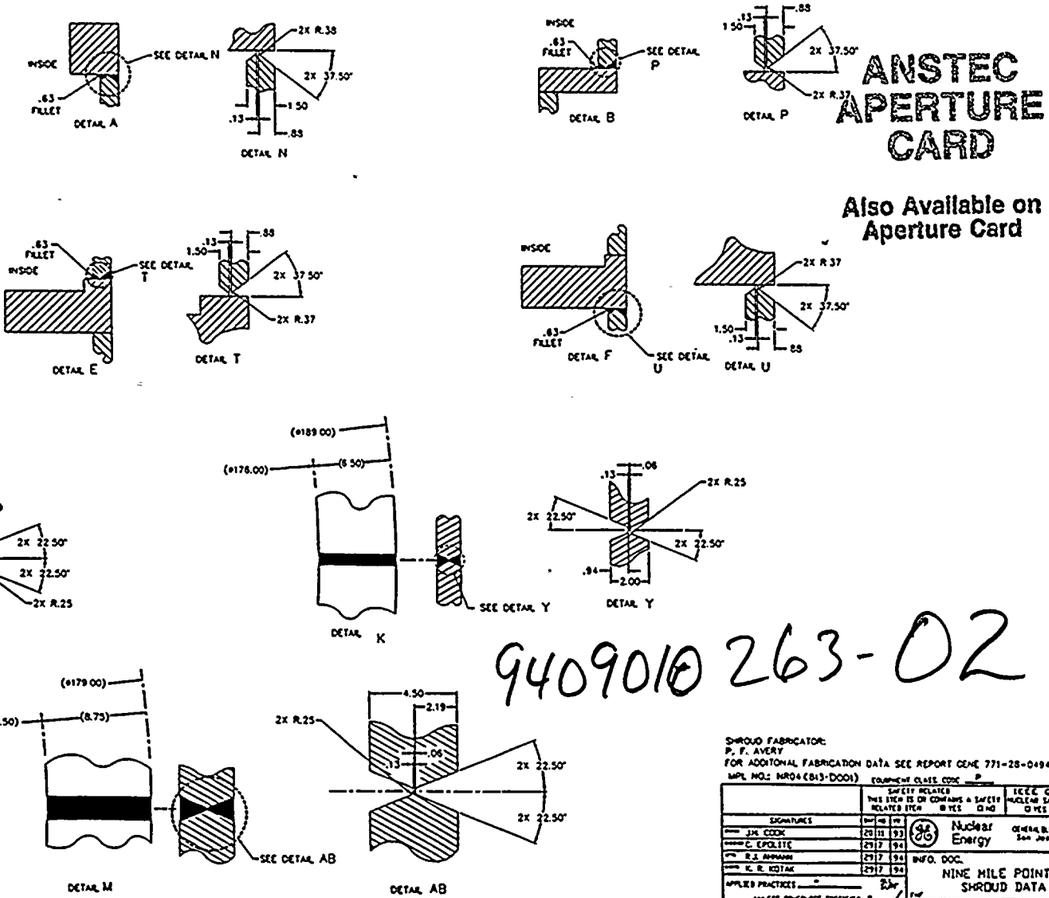
SHROUD SURFACE DEVELOPMENT



PART NUMBER	PART NAME	QUANTITY	MATERIAL TYPE	HEAT NUMBER	CARBON CONTENT - %	COMMENTS ON MATERIAL / PROCESS
1	UPPER RING	2 PIECES	A240 TYPE 304	65444-1	.064	PLATE
2	UPPER CYLINDER	2 PIECES	A240 TYPE 304	65235-1A 65235-1B	.042 .042	
3	CENTRAL RING	2 PIECES	A240 TYPE 304	65294-1	.056	
4	CENTRAL UPPER CYLINDER	2 PIECES	A240 TYPE 304	65235-1	.042	
5	CENTRAL MID CYLINDER	2 PIECES	A240 TYPE 304	65291-1A 840784-2 65290-1 65295-1	.052 .053 .047 .062	
6	CENTRAL LOWER CYLINDER	2 PIECES	A240 TYPE 304	65290-1A 840784-2B 840784-2A 65291-1	.047 .053 .053 .052	
7	LOWER RING	2 PIECES	A240 TYPE 304	65444-1	.064	
8	LOWER CYLINDER	2 PIECES	A240 TYPE 304	65291-1 840784-2A 840784-2B 65290-1A	.052 .053 .053 .047	
9	SHROUD SUPPORT RING	1 PIECE	ASME SA-336 F8	C-23/245352	NOT KNOWN	FORGING

NINE MILE POINT I SHROUD DATA

NOTES:
1. HEAT NUMBER DATA SHOWN MAY BE APPLICABLE FOR NINE MILE POINT 1 AND/OR OYSTER CREEK.



ANSTEC APERTURE CARD

Also Available on Aperture Card

9409010263-02

SHROUD FABRICATOR:
P. F. AVERY
FOR ADDITIONAL FABRICATION DATA SEE REPORT GENE 771-28-0494
MPL NO.: NR04 (843-0001)
EQUIPMENT CLASS CODE P

SIGNATURES			
DRN J.M. COOK	29	11	93
DSN C. EPOLITE	29	7	94
STR R.S. ARMANN	29	7	94
APR K. R. KOTAK	29	7	94

SAFETY RELATED: THIS ITEM IS OR CONTAINS A SAFETY RELATED ITEM ☒ YES ☐ NO
IEEE CLASS 1E NUCLEAR SAFETY RELATED ☐ YES ☒ NO

General Electric Company San Jose CA — Nuclear Energy

INFO. DOC.
NINE MILE POINT I
SHROUD DATA

APPLIED PRACTICES: 2/Ar
UNLESS OTHERWISE SPECIFIED
TOLERANCES ON:
2 PLACE DECIMALS ± = FRACTIONS ± =
3 PLACE DECIMALS ± = ANGLES ± =

105E1413A — REV 0 — SCALE NONE — SHT 1 OF 1
RU-00919 — SIZE C — 2 SKETCH

105E1413A
REV NO. 1 SHT NO. 1
REV NO. 0 SHT NO. 2
REVISIONS