UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING RESULTS OF THE REINSPECTION OF THE CORE SHROUD

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT 1

DOCKET NO. 50-220

1.0 INTRODUCTION

1.1 Purpose

This safety evaluation (SE) assesses the reinspection results and analyses submitted by the Niagara Mohawk Power Corporation (NMPC and the licensee) for its core shroud at Nine Mile Point Nuclear Station, Unit 1 (NMP1) to determine whether the core shroud meets the structural integrity requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI.

The NRC staff considers the core shroud repair by means of shroud stabilizer assemblies (also called tie rod assemblies) to be an alternative to an ASME Code repair under 10 CFR 50.55a(a)(3)(i). Since the assumption for the repair is that the circumferential welds are fully cracked, the vertical welds need a minimum ligament to maintain the structural integrity of the shroud. The NRC staff reviewed the analysis of the vertical welds to determine whether the results supported the alternative repair. The alternative repair and Code requirements are discussed in Section 2.0 of this SE.

1.2 Background

By letters dated January 6 and January 23, 1995, NMPC submitted an application for repairs to the NMP1 core shroud. The shroud repairs and the use of stabilizer assemblies (tie rods) were submitted as an alternative, as discussed above. The NRC staff approved this repair in its SE dated March 31, 1995, according to which NMPC was also to address certain issues by the upcoming spring 1997 inspection (refueling outage 14). NMPC committed to submit its plan for reinspection of the core shroud repair assemblies and the core shroud following refueling outage 13. Also, as noted in the SE, the NRC staff recommended that NMPC perform certain activities to qualify the ultrasonic testing (UT) techniques used to inspect weld H8 and to develop an effective method to locate the segment welds of the top guide support ring. The NRC staff also noted in the SE that NMPC would reinspect all reported indications on the top side of the H8 weld during refueling outage 14. The reinspection would verify the postulated crack growth of these indications.

During the post-installation inspection of the shroud repair, NMPC identified conditions that differed from the intended repair design and reported them in

9705160107

its letters of March 25 and April 30, 1995. NMPC committed to take appropriate corrective actions during refueling outage 14 to restore the shroud stabilizers to their original design. NMPC proposed the final design modifications in its letter of August 14, 1996, as part of the long-term corrective action plan. In its SE of March 3, 1997, the NRC staff reviewed the anomalies in the installed shroud repair hardware and approved the proposed corrective actions.

By letter dated October 4, 1995, NMPC submitted its inspection plan for the core shroud and its repair assemblies. The plan described the inspections projected for the core shroud stabilizer assemblies, the repair anchorage, the H8 weld, the top guide ring segment welds, and the vertical welds. In its letter of February 7, 1997, NMPC confirmed its intent to conduct shroud inspections in the spring 1997 outage according to the inspection plan it previously submitted. NMPC also presented additional information about the inspection plan, addressed the issues in the SE of March 31, 1995, and provided the fracture mechanics analysis that determined the next required ultrasonic inspection of the H8 weld.

In its letter of February 28, 1997, NMPC updated the plans for inspecting the vertical welds described in the October 4, 1995, letter. The inspection scope discussed in the letter was based on a draft version of the Boiling Water Reactor Vessel and Internals Project-07 "Guidelines for Reinspection of BWR Core Shrouds," EPRI Report TR-105747 (BWRVIP-07). NMPC stated that its current plans are in conformance with the BWRVIP-07 guidelines, with one exception related to the expansion criteria for inspection of vertical welds. NMPC also corrected statements made in its letter of February 7, 1997, about the scope of inspection of the H8 weld.

On March 3, 1997, the NRC staff issued its SE of the proposed inspection program. It found that NMPC addressed the issues noted in the SE of March 31, 1995. It also found NMPC's plans acceptable, but stated that the inspections for this outage should be conducted in accordance with the BWRVIP-07 guidelines without the exception. Although the NRC staff has not completed its review of these guidelines, it accepted their use for this NMP1 refueling outage.

NMPC inspected core shroud vertical welds in accordance with the BWRVIP-07 guidelines. On March 20 and April 2, 1997, it notified the NRC staff (by telephone) of the inspection findings. NMPC found vertical weld cracking that exceeded the screening criteria of BWRVIP-07. Additionally, its inspections of the four tie rod assemblies showed that the tie rod nuts had lost some preload (clamping force) and that the lower wedge retainer clips on three tie rods were damaged.

In its submittal of April 8, 1997, supplemented by letters of April 25 (two letters), 26, and 27 (two letters), and 30, 1997, NMPC reported the findings of its inspections. It presented a root cause analysis, corrective actions, design documentation to establish the acceptability of the vertical weld cracking, a weld reinspection schedule, and details of the actions taken to restore the tie rods to the as-designed condition and to modify the lower wedge retainer clip design.

2.0 CODE REQUIREMENT AND ALTERNATIVE

In accordance with ASME Code Section XI, 1983 edition, Summer 1983 Addenda, NMPC chose to perform corrective measures for the cracked circumferential shroud welds on a pre-emptive basis. The repair for cracked welds to satisfy the repair rule of Section XI, IWA-4000 would be to remove the defect and conduct a repair by welding. This is impractical since the core shroud is irradiated to an extent that might cause the shroud to crack further if it were welded. Therefore, as an alternative to IWA-4000, NMPC chose to repair the core shroud with mechanical tie-rod assemblies that serve as a replacement for the circumferential welds.

The NRC staff has approved the tie-rod assembly as an acceptable alternative repair in an SER dated March 31, 1995. The current request, which is the subject of this SE, is to modify a component of the tie-rod assembly (i.e., the lower wedge retainer clip) pursuant to 10 CFR 50.55a(a)(3)(i) and institute a revised installation procedure to ensure that pre-load of the tie-rod assemblies will not be lost by unanticipated motion of the lower toggle bolt supports.

The design assumptions for the alternative repair are that circumferential welds are cracked through-wall for 360° and that the tie-rod assemblies will accommodate the vertical and lateral loads for the "stacked cylinders." The cylindrical sections are fabricated from rolled plates. Two vertical welds are used to join the sections, completing the cylinder. Therefore, vertical weld structural integrity is needed to ensure that the cylinders will remain as cylinders. The crack growth rate used as a part of the fracture mechanics analysis for assessing the needed weld integrity is considered bounding if the plant water chemistry is maintained in accordance with the EPRI BWR Water Chemistry guidelines. However, the NMP-1 current TS is not consistent with the BWR Water Chemistry guidelines. NMPC has committed to submit an application for a license amendment within 60 days to address this matter.

The NRC staff has reviewed the modification and the fracture mechanics analysis that ensure the structural integrity of the vertical welds for the requested operating period of 10,600 hot operating hours and finds that the alternative repair is acceptable pursuant to 10 CFR 50.55a(a)(3)(i). This approval is predicated on the condition that NMP-1 is operated in accordance with the BWR Water Chemistry guidelines, Electric Power Research Institute technical report TR-103515, "BWR Water Chemistry guidelines-1996 Revision."

Details of the review are discussed in the following sections.

3.0 SHROUD INSPECTIONS

Figures 1-1 and 1-2 show the location of the shroud welds.[1] Table 2-2 shows the findings for vertical weld inspections. NMPC did not present a table of findings for the horizontal welds, but provided examination indication maps

[1] All figures and tables attached to this NRC staff SE are extracted, without change, from reports submitted by NMPC. Therefore, figure and table numbers do not necessarily correspond to the section numbering within this SE.

and discussed the results in a qualitative way in Appendix C of the report by General Electric Nuclear Energy (GENE), titled "Assessment of the Vertical Weld Cracking on the NMP1 Shroud," GENE-523-B13-01869-043 (portions attached). The scope, methods, and findings of the inspections are discussed hereinafter in Sections 3.1, 3.2, and 3.3, respectively.

3.1 Inspection Scope

As discussed in Section 1.0, NMPC stated that its current plans were in accordance with the BWRVIP-07 guidelines, but with one exception. The exception concerned the inspection of vertical welds. For the vertical weld inspections scheduled for the spring 1997 outage, NMPC proposed to modify the method for sample expansion presented in BWRVIP-07 as Option B.

According to Option B, if the cumulative cracking in either the original sample or the expansion sample exceeds 10 percent of the equivalent length of weld inspected, the inspection scope is to be expanded to verify the minimum required uncracked length for each vertical weld that is not structurally replaced by existing hardware or the repair or both.

NMPC proposed to expand the inspection scope to verify the minimum required uncracked length for each vertical weld not structurally replaced by existing hardware or the repair or both only after finding the cumulative cracking in either the 50-percent expansion sample or the 100-percent expansion sample greater than 10 percent of the equivalent length of weld inspected.

The NRC staff did not grant the exception, finding that NMPC did not present adequate technical bases to deviate from the BWRVIP-07 guidelines. The NRC staff's judgment was that the provisions of Option B should be followed and hence, a finding of more than 10 percent of the equivalent length of weld inspected being cracked would merit additional inspections to further qualify the degree of cracking.

NMPC inspected the NMP1 vertical welds according to the sampling option of BWRVIP-07. This option specified a visual inspection of 25 percent of the equivalent total vertical weld length from either the outside diameter (OD) or inside diameter (ID).

The ring segment welds (labeled in Figure 1-2 as V5 and V6) were excluded from the vertical welds requiring inspection based on a GE analysis of the ring segment welds submitted to the NRC staff for review by letter dated February 7, 1997.

The initial inspection of the vertical welds with enhanced visual techniques found cracking over the entire OD length of the V10 weld. NMPC then expanded the inspection plans to establish the minimum required uncracked ligaments on the vertical welds that are required to meet the shroud stabilizer repair design-basis assumptions. It performed the inspection using an enhanced visual inspection (EVT) method supplemented by UT.

NMPC found extensive cracking on the OD of vertical welds V9 and V10. As a result, NMPC performed a complete baseline inspection of the accessible parts

of most horizontal and vertical welds to assess the overall material condition of the NMP1 core shroud. NMPC examined the accessible areas of the ring but was unable to locate the welds V5 and V6. NMPC inspected the horizontal welds H2, H4, H5, H6a, H6b, and H7, even though inspection of horizontal welds was not required because a preemptive repair had been installed. Such repairs are designed assuming the circumferential welds are fully cracked (through-wall for 360 degrees), obviating the need to inspect them.

The inspection scope is summarized in Table 2-2 and Appendix C. The scope was sufficiently comprehensive to identify degradation that could invalidate design assumptions for the tie rod repairs. In expanding the scope of inspections, and inspecting from the ID and the OD of the weld, NMPC determined whether each vertical weld contained the minimum ligament needed for maintaining the structural integrity of the shroud.

The relation of the results of vertical weld inspections to the design assumptions for the tie rod repairs is discussed previously in Section 2.0 of this SE.

3.2 Inspection Methods

Table 2-2 and Appendix C indicate the inspection method used for the various welds. EVTs are performed with a video camera qualified in tests to detect a 0.5-mil wire. Visual inspections cannot determine the depth of a flaw. They are used to detect surface flaws on either the ID or OD. Ultrasonic inspections are volumetric examinations used to detect flaws and determine their size. Accessibility of the weld determines which method is used. Table 2-2 shows that each of the vertical welds was examined from the OD and ID either by EVT or UT or both. Supplemental eddy current inspections were also used to determine if defects were present on the ring segments containing welds V5 and V6. These welds are difficult to locate by UT or visually because the support ring was machined after fabrication and welding.
The UT for examining the core shroud at NMP1 was performed with a GE delivery device, the suction cup scanner. Three transducers were used including a 45° shear, 60° longitudinal, and an OD creeping wave. The inspection device was demonstrated and qualified to detect and determine the size of cracks on a core shroud mock-up at the EPRI Nondestructive Examination (NDE) Center. This device has been used to perform core shroud inspections at a number of other BWRs.

The crack size (length and depth) measurement uncertainties for this device were determined in accordance with guidelines in the "Reactor Pressure Vessel and Internals Examinations Guidelines (BWRVIP-03)" (EPRI TR-105696), October 1995. In measuring length, uncertainties arise from the delivery system and the NDE technique. The qualification of the system at the EPRI NDE Center included quantification of measurement uncertainties. The uncertainty factors from the delivery system and the NDE technique were determined to be 1.106 inches and 0.364 inches, respectively. The combined uncertainty factor of 2.94 inches [(1.106 inch + 0.364 inch) x (2 ends)] was applied to each ultrasonically measured flaw length including the calculated critical crack length. The 60° longitudinal wave transducer was used in depth measurement. The uncertainty factor associated with the depth measurement was determined to

be 0.108 inch (no contribution from delivery system), and was applied to each UT depth measurement.

The EVT performed for the core shroud inspections was qualified by demonstrating the capability of resolving a 0.5-mil-diameter wire. The uncertainty factor for the length measurement by the EVT was determined to be 1.2 inch. To account for the measurement uncertainty, 2.4 inches (for two ends) were added to each visually measured flaw length.

3.3 Inspection Findings

NMPC presented detailed descriptions and examination indication maps of the vertical weld cracking. The results are summarized in Table 2-2 and are included in Appendix C (attached). The inspections showed significant cracking on welds V4, V9, and V10. Minor cracking was found on welds V3, V12, V15, and V16. No cracking was found on the accessible areas of welds V7, V8, and V11. Welds V5 and V6 were not located.

The weld with the worst cracking was V9. It had numerous indications on more than 90 percent of the weld and up to 80 percent of the depth. Weld V10 had numerous indications on more than 80 percent of the weld.

NMPC tried to examine the ring segment welds V5 and V6, welds known to be difficult to examine given current technology. Using eddy current and visual inspections, NMPC could not locate the specific welds. However, EVT of the accessible outer surface of the ring segment did not find any cracking.

The NRC staff approved NMPC's inspection plans for these welds in its SE of March 3, 1997. Those plans contained an analysis that showed that if NMPC could not locate these welds, the shroud repair would remain effective even if the welds were significantly cracked. NMPC's evaluation showed that the structural integrity of the ring would be maintained even with cracking up to 95 percent of the segment welds. However, the tie rod repair may be less effective if ring segment welds are totally cracked and the two neighboring horizontal welds, H2 and H3, are also completely cracked. Under this condition, the stiffness of the core shroud may change and, consequently, affect the amount of the thermal preload applied by the tie rod assemblies during normal operation. The inspections verified that sufficient intact weld length exists in the H3 weld to support operation, even if the ring segment welds were significantly cracked.

NMPC submitted indication maps of the horizontal welds and a qualitative description rather than a table of quantitative results. The installed stabilizer assemblies were designed to ensure shroud integrity even if all the horizontal shroud welds were cracked through the wall around the full circumference of the shroud. Thus, the condition of the horizontal shroud welds did not need a quantitative assessment to support continued operation. The findings of the inspections are, however, included in Appendix C (attached).

The structural significance of the indications is assessed in Section 4.2 and the potential for bypass leakage in Section 4.3.

4.0 EVALUATION OF INSPECTION FINDINGS

4.1 Root Cause

NMPC determined that the cracking in the vertical welds was caused by intergranular stress corrosion cracking (IGSCC). NMPC's findings agree with research and operating experience with this phenomenon.

It is well known that the core shrouds of all boiling water reactors (BWRs) are susceptible to IGSCC. The following relevant factors affect the cracking: operating time, coolant conductivity, material carbon content, plate orientation, fabrication-related surface cold work, neutron fluence, residual stresses resulting from welding and fabrication, and operating stresses. The NMP1 shroud is susceptible to IGSCC because it is made of high-carbon Type 304 stainless steel that is sensitized by welding and subjected to residual stresses from welding and fabrication. NMP1 is classified by industry guidelines as a Category C plant, a category that contains the most susceptible core shrouds. Category C plants have core shrouds made of Type 304 stainless steel and more than 6 hot operating years, regardless of reactor coolant conductivity.

The location of the cracking in the NMP1 shroud is consistent with IGSCC as shown by inspection data. The cracking was located at heat-affected zones (HAZs) sensitized by welding and in areas with residual stresses in which fabrication-related welding or grinding was apparent. In a few limited instances, cracking was found to extend up to 1.5 inches away from the weld perpendicular to the longitudinal axis of the weld. A few crack indications not associated with HAZs were found in the base metal. However, these cracks initiated from the cold-worked areas resulting from grinding to remove the attachment welds (lugs) during construction.

The cracks are expected to stay confined to the HAZs or in the severely cold-worked area. Radiation, when exceeding a threshold value, can sensitize materials, making them susceptible to IGSCC. However, offsetting this effect, is the reduction of tensile residual stresses outside the affected areas. Unless high tensile stresses, such as those that result from grinding or cold work are present, there is not enough driving force to propagate a crack beyond the affected areas. The depth of the cold-worked layer in the base metal resulting from grinding is generally very shallow and, therefore, the cracking in the cold-worked areas will also not grow very deep.

NMPC contended that although some cracks extended beyond the HAZ, they did not differ from those seen in other BWR plants in the past. To check the contention, the NRC staff reviewed inspection data for circumferential welds, considering data for the H5 weld at Brunswick Unit 1 as an example. The inspection report for that plant of June 21, 1995 (Brunswick Steam Electric Plant No. 1, Docket No. 50-325/License No. DPR-71 NRC Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," letter dated June 21, 1995, from R. Anderson of Carolina Power and Light Co. to the NRC) for the 10th refueling outage states that H5 was visually examined 100 percent of the ID and 45 percent of the OD. Five of the cracks on the OD were circumferential, 0.5 to 3 inches long. The remaining

cracks were axial, extending less than 9 inches from the toe of the weld. Long circumferential cracks and short axial cracks, less than 4.5 inches long, were on the ID. Baseline depth measurements were taken by UT in two locations to benchmark future inspections. The depth ranged from less than 0.3 inches to 0.6 inches.

That inspection report further states that during the 10th refueling outage, "punch marked cracks" on the ID were visually reinspected for length. In addition, two areas inspected by UT during the 9th refueling outage were ultrasonically reinspected to determine crack growth and were compared to previous sizing data. The report concluded that there was no change in depth or length. This conclusion supports the contention that cracks are not likely to propagate outside the cold-worked area and that the crack growth rate is slower than the NRC's bounding rate of 5×10^{-5} inch per hour.

The NRC staff considered the question of the adequacy of base metal inspections. HAZs are typically narrow and adjacent to the weld. NMPC reported that the HAZs associated with the vertical welds near the beltline extended about 0.8 to 1.0 inches from the weld fusion line. NMPC visually examined about 2.5 inches of base metal on each side of each vertical weld. Considering the extent of the examination together with the fact that the driving force for cracks is greatly reduced outside the HAZ, the NRC staff concluded that NMPC examined enough area outside the HAZ to determine the condition of the base metal.

The NRC staff considered the effect of radiation fluence on the cracking observed. Irradiation increases the susceptibility to IGSCC, and the fluence was in a range for which irradiation-assisted stress corrosion cracking (IASCC) conditions can occur. The estimated fluence at welds V9 and V10 was in the range of 2 to 4.5×10^{20} n/cm^2 (>1 MEV). NMPC will check whether IASCC was a contributing mechanism to the cracking by analyzing boat samples of cracked material. This work will be completed and the results will be submitted to the NRC for review. This issue does not need to be resolved immediately because the crack growth rate used takes into account the effect of fluence. Crack growth rates are evaluated in Section 4.2.

The NRC staff finds that NMPC's analyses reasonably explain the observations; they showed that the welding and fabrication processes caused the cracking pattern seen on the vertical welds. They also showed that the pattern of crack depth is consistent with the calculated fluence axial and radial profiles. That is, both welds V9 and V10, the welds with the worst cracks, are between welds H4 and H5 and are exposed to neutron irradiation along their entire length. The axial cracks in weld V9 were found to be the deepest near the H4 weld where the fast neutron flux is highest.

4.2 Structural Integrity Assessment

The methodology for assessing the structural integrity of a flawed core shroud axial weld, is broken down into a number of steps. First, the size of the flaw at the time of the analysis (the initial flaw size, a_i) is determined by ultrasonic or visual inspection of the weld and this flaw is conservatively assumed to be present completely through the thickness of the core shroud

wall. This value, a_i, represents the best estimate of the flaw size. However, some uncertainty exists in a_i because of the method used to determine the flaw size. When the uncertainty due to the inspection methodology (discussed in detail in Section 2.0) is added to the best estimate of the initial flaw size, a new value, a_0, is determined. This value, a_0, is taken as the initial flaw size in the structural analysis since it represents a bounding value for the size of the existing flaw. Then a crack growth rate (da/dt) is determined. For this analysis, a bounding value of the IGSCC growth rate was assumed (5×10^{-5} inch per hour) and multiplied by the amount of operating time to be justified (t) to determine a bounding value for the flaw size at the end of the proposed operating time (the final flaw size, a_f). It is this final flaw size, a_f, that must be shown to be less than some critical flaw size determined by prescribed loading conditions (with appropriate safety factors) according to structural integrity assessment methodologies.

The flaw sizes based on reported measured values and the uncertainties associated with these measurements are discussed in Section 3.3. Section 4.1 discusses assumed crack growth rates. Section 4.2 discusses structural integrity assessment methodologies. Section 4.3 summarizes NMPC's structural integrity assessment, and Section 4.4 discusses the NRC staff's independent analyses.

4.2.1 Evaluation of Crack Growth Rates

GE, in its analysis for NMPC, assumed a bounding crack growth rate of 5×10^{-5} inch per hour. This crack growth rate bounds laboratory test data for a variety of water chemistries, bounds field experience with IGSCC of other components of the reactor coolant system, and has been used by the NRC for flaw growth evaluations. Even after considering potential IASCC contributions, the actual crack growth is expected to be lower than this bounding value.

GE predicted the crack growth rate for the vertical welds using several methods. It considered the effects of radiation on crack growth rates, comparing predictions of rates for unirradiated material to those for irradiated materials at the same values of reactor water conductivity, electrochemical potential (ECP), and initial sensitization. The comparison showed that at the fluence levels at BWR shrouds, the predicted crack growth rate is similar for both the unirradiated and the irradiated materials. GE attributed the similarity in rates to offsetting effects. Although irradiation increases the susceptibility of the material, it also relaxes the weld residual stresses, the driving force for crack growth. Ultimately, GE concluded that the crack growth rate over the range of shroud fluences is bounded by the NRC bounding crack growth rate of 5×10^{-5} inch per hour. The NRC staff agrees with GE's determination.

The NRC staff considered the effect of the distribution of residual stresses on the crack growth rate. Similar results to those reported by MPM Technologies in its "Analysis of Nine Mile Point Unit 1 Shroud Welds V9 and Weld V10 Cracking," Report No. MPM-497439, April 1997, were obtained by researchers performing NRC-sponsored work at Argonne National Laboratory

(ANL). Researchers at ANL analyzed the distribution of residual stresses from welding in shroud welds. The work at ANL, unlike that of MPM, did not include residual stresses induced by fabrication, but did find residual stresses from welding to be highly tensile at the surfaces and compressive in the center of the wall of the cylinder. Additional observations include that actual conditions like "out of roundness" would bias cracking to a particular surface. Further, if stresses were uniform across the thickness, cracks would grow at about a 2:1 aspect ratio. Even assuming multiple crack initiation sites, the aspect ratios found in the cracking at NMP1 imply that the stresses decrease sufficiently through the wall to reduce the stress intensity. The decrease in stress intensity is clearly more than sufficient to show that 5 x 10^{-5} inch per hour is a conservative rate of crack growth.

The NRC staff examined the question as to whether crack growth rates in vertical welds differ from the rate for horizontal welds. The rates are expected to be similar for the following reasons: The cracking on both the horizontal and vertical welds is caused by the same mechanism, IGSCC, in analogous locations, that is in weld HAZs, which are areas of high residual stress. The residual stress patterns are similar for both kinds of welds. They are tensile on the outer surfaces and decrease, even become compressive, within the wall and with distance away from the weld.
At NMP1, no field data on crack growth are available for vertical welds. Of the horizontal welds, field data are available only for the H8 weld. During the current outage, NMPC located a flaw found by UT during the previous (13th) refueling outage as well as an additional flaw in the same area. Through UT, NMPC determined that the flaw was of less through-wall depth than in the 13th refueling outage. NMPC concluded, after reviewing the previous data, that the earlier sizing was very conservative. NMPC drew no conclusions from these data about the crack growth rate at NMP1, and the bounding rate of 5 x 10^{-5} inch per hour was used in NMPC's analyses.

The NRC staff examined the question of whether the water chemistry at NMP1 conforms to the requirements under which the bounding rate is applicable. This rate was derived from laboratory test data. The reference for the data is a letter from NMPC to NRC, "Responses to NRC Staff Questions Provided During Telephone Conversations of April 22, 1997, on Core Shroud Cracks and Repair," dated April 27, 1997. The crack growth tests for sensitized stainless steel were performed at a range of conductivities from 0.3 to 1.5 microSiemen/cm. The data from these tests are bounded by the crack growth rate of 5 x 10^{-5} inch per hour.

In its letter of April 30, 1997, NMPC committed to continue operating in accordance with the EPRI Water Chemistry Guidelines. These guidelines are also incorporated into BWRVIP-07; that document assumes that plants meet the requirements specified by the EPRI Water Chemistry Guidelines Action Level 1. According to Action Level 1, if the conductivity exceeds the limit of 0.3 microSiemen/cm, the conductivity must be reduced to that value or lower within 96 hours. Although the conductivity limit in NMP1 Technical Specification 3.2.3 is 5 microSiemen/cm (when steaming rate are equal to or greater than 100,000 pounds per hour), NMP1 has been operating at a much lower conductivity, less than 0.10 microSiemen/cm, during the past three cycles and less than 0.3 microSiemen/cm during the last seven cycles. Therefore, the

bounding crack growth rate applies to NMP1 because this rate is conservative with respect to Action Level 1 in the EPRI Guidelines. By letter dated May 7, 1997, NMPC has committed to submit a license amendment within 60 days that will address the difference between the current reactor coolant system chemistry requirements of NMP1 TS 3.2.3 and the coolant chemistry criteria referenced for core shroud crack growth rates as described in NMPC's letter of April 8, 1997. The NRC staff finds this commitment acceptable.

4.2.2 Structural Integrity Assessment by Linear Elastic Fracture Mechanics and Limit Load Analysis Methodologies

Either of two analysis methodologies, limit load analysis (LLA) or linear elastic fracture mechanics (LEFM), may be applied to demonstrate the structural integrity of flaws in ductile materials such as Type 304 stainless steel. Either of these methods seeks to determine the length of flaw that can be tolerated (or, conversely, the amount of uncracked ligament that must remain) for the location to support the applied stresses under prescribed loading conditions. These loading conditions are divided for the purpose of this analysis (and by the ASME Code) into two general categories: (1) normal operation or upset conditions and (2) emergency or faulted conditions. The significance of this is that the ASME Code analysis procedures require that a safety factor of 3.0 be applied to the loads determined for normal operation or upset conditions, while a safety factor of 1.5 is to be applied for emergency or faulted conditions.

The first methodology, LLA, determines the amount of uncracked ligament that must remain to keep the location from being pulled apart by full-section plastic yielding. An LLA is usually applicable to materials that retain a high level of fracture toughness and demonstrate no crack extension under loadings up to those that cause plastic yielding. The results of this method are then compared to the results of an LEFM analysis to determine which method controls the evaluation (i.e., gives the smallest acceptable flaw size).

LEFM is a conservative method for assessing the potential for crack growth in a ductile material. LEFM assumes that the material will behave in a nonductile fashion (which reduces its flaw tolerance) and compares the stress intensity due to the flaw and the applied loading to the material's fracture toughness (a quantitative measure of the material's flaw tolerance). LEFM would normally be expected to be appropriate for materials that have lost ductility, for example, from irradiation damage.

4.2.3 NMPC's Evaluation of Structural Integrity

GE performed the analysis that determined the structural margin of the vertical welds.

The assumptions for the shroud repair are that horizontal welds are cracked through-wall. Under this assumption, vertical weld cracks are acceptable, as long as the crack lengths are less than the allowable flaw sizes or as long as the structural integrity of the vertical weld can be shown. Typically, the allowable crack sizes are large and approach or exceed the length of the weld itself.

The analysis determined that the primary stress that could cause vertical weld failure would result from the internal pressure. Consistent with ASME Code practice (Appendix C, Section XI), the analysis considered internal pressure as the only load for axial cracks. It considered the internal pressures under all conditions--normal, upset, and accident events--with the appropriate ASME Code safety factors.

The analysis did not take credit for the horizontal weld integrity in determining the allowable vertical weld flaw sizes. The horizontal welds are assumed to be cracked completely through the wall, and the cylinder between any two horizontal welds is assumed to be "stand-alone." The calculations also assumed simultaneous cracks at the diametrically opposite welds in a given cylinder. The analysis applied both LEFM and limit load to determine the allowable flaw sizes. Calculations for the required minimum ligament allowed for crack growth and inspection uncertainty.

In the analysis, GE first determined whether the vertical welds met the screening criteria specified in BWRVIP-07. GE then performed a more detailed fracture mechanics analysis to demonstrate the integrity of vertical welds with indications that did not conform to the screening criteria.

Allowable shroud vertical crack lengths were calculated on the basis of both LEFM and LLA. The high fluence region in which there is potential irradiation embrittlement is limited to the shroud section between horizontal welds H3 and H6a. Therefore, both LEFM and LLA were used for vertical welds in this region. LEFM was found to be governing for welds V9, V10, V11, and V12, and LLA was governing for welds V7 and V8. The vertical welds in other regions were governed by LLA.

Table 5-2, Columns 3 and 4, (see enclosure to this SE) shows the allowable crack length as well as the required uncracked ligament lengths for each vertical weld. In accordance with BWRVIP-07, allowances for crack growth and uncertainty in the inspection were added to the required uncracked ligaments (Column 5 of Table 5-2).

Cracking of radial ring welds was separately evaluated and found to have negligible impact. The only effect of radial ring weld cracking is on the thermal preload, but the analysis found that even with 90 percent of the weld assumed to be cracked, the effect on preload was insignificant.

Column 6 in Table 5-2 lists the values of uncracked ligament lengths determined from the UT and EVT of the various shroud vertical welds. A comparison of these lengths with those required by the conservative screening criteria lengths (Column 5) shows that each of the vertical welds except for V4, V9, and V10 meets the required uncracked ligament length criteria.

The analysis showed that each weld except for V4, V9, and V10 is acceptable for continued operation for at least a fuel cycle of 16,000 hours. These three welds were analyzed further. In this more refined analysis, credit was taken for uncracked ligaments for part-through cracks after accounting for crack growth.

Of the three welds, V9 was determined to be the most limiting case and was, therefore, evaluated first. The analysis applied the required safety margins consistent with the ASME Code Section XI criteria for piping. The analysis justified continued operation for at least 10,600 hours (approximately 14-1/2 months).

Weld V4 was evaluated by LLA because it is located above the top guide support ring where the fluence is low enough that the material is not embrittled. Table 5-2 shows the required uncracked ligament. The analysis showed that the limit load margins are satisfied for the V4 weld for a period of at least 10,600 hours.

The structural analysis concluded that continued operation at NMP1 is justified for an additional 10,600 hours (about 14-1/2 months). The allowable hours of operation ensure that the required ASME Code margins are maintained, considering both LEFM and LLA. The NRC staff finds this conclusion acceptable. The analysis was performed according to methodology approved by the NRC staff. The methodology is detailed in the General Electric Nuclear Energy document GENE-523-113-0894, Revision 1, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," March 1995, and has been used for circumferential welds. It is incorporated by reference in BWRVIP-07.

4.2.4 Independent NRC Staff Structural Integrity Analyses

The NRC staff has performed an independent, quantitative assessment of the largest axial flaw in the NMP1 core shroud. The NRC staff evaluated this flaw using both LLA and LEFM techniques. The NRC staff found that the LEFM analysis, presented below, controls the evaluation.

The NRC staff accepted NMPC's best estimate for the largest initial flaw length of approximately 74 inches. The NRC staff accounted for the uncertainty in the sizing of the initial flaw by adding the uncertainty factor in GE's ultrasonic testing method (1.5 inch) to each end of the best-estimate flaw. Therefore, the NRC staff used a value of 77 inches for a_0, the bounding initial flaw size. The NRC staff then applied the bounding crack growth rate (5×10^{-5} inch per hour) for 10,600 hours (approximately 14-1/2 months) to determine a bounding final crack length.

The NRC staff then applied the stresses that were reported to result from faulted loading conditions with the safety factor of 1.5 (as these were reported to be the most challenging loading conditions) and used an analytical equation that contained a small correction factor for long cracks that was not included in GE's LEFM analysis. This resulted in a calculated stress intensity of about 175 ksi$\sqrt{\text{in}}$ at the tip of the bounding final crack length. The NRC staff compared the result with the fracture toughness of 150 ksi$\sqrt{\text{in}}$ reported by GE from test results for material removed from a foreign reactor. Further, the NRC staff has determined that additional test data for weld material from another foreign nuclear power plant indicate similar fracture toughness to that discussed. In addition, an elastic/plastic fracture analysis was performed by Structural Integrity Associates with the fracture toughness of the base metal reduced by about a factor of 3 to bound weld metal properties. The analysis by Structural Integrity Associates produced results

that were similar to the LEFM analysis performed by GE. On the basis of the information reported by NMPC and the NRC staff's analysis, the bounding final crack length for the limiting core shroud flaw is slightly larger than would be acceptable under LEFM analysis. This independent evaluation by the NRC staff used a handbook solution that is not as precise as NMPC's finite element analysis. The results from the analysis is that the assumed material toughness is slightly exceeded. Considering the test data from the two foreign nuclear power plants yielding similar values for fracture toughness, the results of the elastic/plastic analysis performed by Structural Integrity Associates and the uncracked ligaments for which credit was not taken in NMPC's analysis, the NRC staff concludes that the more refined analysis performed by NMPC appears reasonable. Finally, the NRC staff confirmed that the LEFM analysis was controlling by determining the acceptable amount of remaining ligament according to LLA to be 7.0 inches, much less than that required by the LEFM analysis. Furthermore, evidence exists that the materials removed from shrouds that have been tested are failing in a ductile manner, indicating that LLA or elastic/plastic analysis may be more appropriate analysis methods.

4.3 Assessment of Potential for Bypass Leakage

NMPC evaluated the potential crack opening and associated leakage for the V-9 and V-10 vertical welds, as documented in General Electric Nuclear Energy Report GENE-B13-01869-043, Revision 0, "Assessment of the Vertical Weld Cracking on the NMP1 Shroud," April 1997. The evaluation also considered whether the leakage would be detectable. The NRC staff's review of NMPC's leakage evaluation is presented below.

4.3.1 Leakage Estimate

By conservatively assuming that both the V-9 and V-10 welds were cracked through-wall, NMPC estimated that the postulated vertical weld cracks would provide 3-square inches of leakage flow area (approximately a 0.003-inch opening along the entire length of the welds). On the basis of this leakage flow area, the total calculated leakage from the V-9 and V-10 welds was estimated to be 200 gpm at 100-percent rated power and 100-percent rated flow. This statistic is equivalent to approximately 0.11 percent of the core mass flow.

In a letter dated March 31, 1995, the NRC staff presented its evaluation of the estimated leakage with the tie-rod assemblies installed and with the postulation of through-wall cracking of all the horizontal welds. The NRC staff found that with the tie-rod assemblies installed, the estimated leakage was 0.33 percent of core flow at 100-percent rated power and 85- to 100-percent rated core flow. The NRC staff notes that the postulated 360° through-wall cracks in the horizontal welds with the tie-rod assemblies installed yield a higher estimated leakage than the total estimated leakage from the V-9 and V-10 welds cracked through-wall. NMPC estimated that the total leakage resulting from the tie-rod assemblies installed and the postulated through-wall cracking of the horizontal welds and the V9 and V10 welds was 0.65 percent of the core mass flow at 100-percent rated power and 100-percent rated core flow. The NRC staff does not consider this a

significant leakage rate since it is less than 1 percent of the total core flow and is, therefore, acceptable.

The NMP1 emergency core cooling system (ECCS) consists of the single-train feedwater coolant injection (FWCI) system, the automatic depressurization system (ADS), and the two-train core spray (CS) system. The FWCI system requires limited offsite power to be functional. During a postulated loss-of-coolant accident (LOCA), the CS system transfers water from the suppression pool to the reactor pressure vessel (RPV) where the water is sprayed upon and cools the core and returns to the suppression chamber by way of the break. CS system water is introduced at the top of the core using circular spargers located inside the core shroud and containing distribution nozzles pointed radially inward and downward to produce fine spray droplets directed onto all fuel assemblies. On the basis of this description of the CS system, the NRC staff notes that the estimated total leakage from the installation of the tie-rod assemblies and the postulated through-wall cracks in the horizontal welds and the V9 and V10 welds does not affect the performance of the CS system. Because NMP1 is a BWR-2, it does not have jet pumps and does not depend upon reflood for a postulated recirculation line or steam line break. Thus, leakage of the shroud is not an issue for a recirculation line or steam line break since the core would be cooled by spray cooling. Therefore, ECCS performance is not affected by the increase in postulated leakage as a result of the V9 and V10 welds.

4.3.2 Detection

NMPC evaluated whether the postulated through-wall cracking of the V9 and V10 welds would be detected as a result of changes in reactor operating parameters. The nuclear industry has established that shroud cracks may be detected if the estimated leakage produces a power anomaly of 2 percent in the rated power. The NRC staff notes that approximately 3-percent leakage is required to produce a 2-percent power reduction. For the estimated leakages discussed above, the combined leakage from (1) the installation holes from the tie-rod assemblies, (2) the postulated cracking in the horizontal welds, and (3) the postulated cracking of the V9 and V10 welds would not be detectable. Any change to other leakage indicators, such as recirculation loop temperature and core support plate pressure differential, would also be too small for the reactor operators to detect.

NMPC postulated that if vertical welds V-9 and V-10 did develop a larger leakage flow area, i.e., a crack opening of 1.5 inch for the entire length of the weld, the power decrease would be more than 2 percent and, therefore, would be detectable. NMPC maintains a Special Operating Procedure N1-SOP-2, Revision 5, "Unexplained Reactor Power Change" that was revised in 1994 to account for potential shroud cracking or displacement. In this procedure, depending upon where the indication is detected (i.e., below the core plate, between the core plate and the top guide, or above the top guide), the operator is instructed to either scram the reactor or commence a normal reactor shutdown. At the NRC's public meeting on April 14, 1997, NMPC stated that operators were retrained on this procedure every 2 years.

On the basis of the estimated leakage rates and the above information, the NRC staff concludes that shroud cracking in the horizontal or vertical welds would not produce effects that could be detected. However, if the crack openings were of significant size along the entire length of the welds (0.25 inches for all horizontal welds or 1.5 inches for both the V-9 and V-10 welds), the NRC staff concludes that these cracks would be detectable, and NMPC has suitable operating procedures in place to shut the reactor down.

5.0 SHROUD STABILIZER ASSEMBLIES

5.1 Summary Description of the 1995 Shroud Repair

In its SE dated March 31, 1995, the NRC staff approved NMPC's submittal of the shroud repair. A summary description of the repair is given in the following paragraphs.

The shroud repair consists of four stabilizer assemblies (also called tie rod assemblies) placed at approximately uniform intervals around the shroud (azimuths 90°, 166°, 270°; and 350°). The stabilizer assemblies were designed to structurally replace the circumferential welds in the core shroud. Each assembly functions to vertically hold the shroud to the shroud support cone and to horizontally support the shroud at the top guide and core plate elevations. There are also other horizontal supports that would prevent unacceptable horizontal movement of any shroud cylindrical segment in the event the horizontal shroud welds should fail.

The stabilizer assemblies provide vertical restraint and represent an alternate load path between the top of the shroud and the shroud support cone. This load path consists of the upper support, tie rod, C-spring, lower support, and toggle bolt. Differential thermal expansion from the different materials used for components of this load path, produces a thermal preload at plant operating conditions. Under postulated failures of the horizontal shroud welds, the thermal preload is sufficient to hold the cylindrical segments of the shroud in place for all normal operating conditions, such as the vertical upward force applied to the shroud by the coolant flow and pressure. The vertical load path is also designed to have a vertical spring rate that both prevents unacceptable vertical load during plant upset thermal conditions and provides acceptable dynamic response during a plant seismic event.

The shroud is restrained horizontally by linear springs at the top guide and the core plate elevations. At the top guide elevation, the linear spring consists of the upper spring, upper wedge, upper contact, and the upper support. At the core plate elevation, the spring consists of the lower wedge, lower contact, and lower spring. The horizontal spring rates of these springs were designed to produce acceptable horizontal dynamic response during a seismic event. The horizontal displacement of the shroud during all events must be limited by these springs to ensure control rod insertion and prevent unacceptable leakage. In addition, in the event of failure of the horizontal shroud welds, unacceptable displacement of other cylindrical sections of the shroud would be prevented by displacement limiters, such as the mid-support and top support.

During the initial installation of the repair hardware, the attachment features were machined into the shroud head and the shroud support cone. The lower support toggle bolts were inserted through the shroud support cone and tightened to 40 foot-pounds. In each assembly, the tie rod nut at the top was also tightened to a specified amount. The lower wedge was machined to provide a 0.01-inch compression of the lower spring.

The mid-support was machined to provide a 0.25-inch horizontal displacement of the tie rod.

The lower wedge is a component that was machined on the basis of actual site measurement to fit between the RPV and the lower spring with a small (0.010 inch) compression of the lower spring at room temperature. The lower wedge is held in place by a latch device. The latch (also referred to as a retainer clip) is a wishbone-shaped piece that is intended to prevent relative motion between the lower wedge and the lower spring. Similar latches are also used to prevent relative motion at the mid-support and at the upper spring. The lower support is an assembly that connects the shroud repair hardware to the shroud support cone. The upper spring jacking bolt was adjusted to slightly compress the upper spring. All moveable features were locked in place with mechanical devices such as crimps or spring retainers. The lower spring wedge latches are one type of spring retainer. Thus, it was intended that, after installation, the upper and lower springs would be compressed between the shroud, and the RPV and the tie rod would be tight with a small preload. The lower support and toggle bolt assemblies were tightened to the shroud support cone, but (as would be discovered during the spring 1997 outage) their position within the support cone holes had not been properly specified in the installation procedures. They could be at any location within the hole depending on the hole's size and shape.

As stated earlier, the NRC staff reviewed NMPC's submittal relating to the shroud repair hardware and found it acceptable.

5.2 NRC Staff Evaluation of the Stabilizer Assemblies

5.2.1 Root Cause Evaluation of the Shroud Repair Deficiencies and Corrective Actions

During the spring 1997 outage, NMPC performed post-operational inspections on the core shroud stabilizer (tie rod) assemblies, consistent with its commitments and NRC staff requirements. Deficiencies were found in the tie rod repairs, which included, to varying degrees, loose tie rod nuts on all four tie rods, damage to the retainer clips on the lower spring wedges, and mispositioned mid-supports. NMPC performed root cause evaluations, additional inspections, and testing of the tie rods. NMPC described these matters to the NRC staff in a letter dated April 8, 1997, forwarding GE's analysis of the repair.

The tie rod nut at the 270° location was first discovered to be loose. The nut-locking device was normal and, in accordance with the original design and installation requirements, the nut could not be moved without removing the locking feature. However, there was essentially no preload between the nut and the tie rod. After the locking feature was removed, the nut was turned with less than 25 foot-pounds of torque. The rotation of the nut before tightening it at 25 foot-pounds was equivalent to an axial clearance of approximately 0.1 inch between the nut and the tie rod. NMPC's inspection and analysis confirmed that the nut had loosened because the toggle bolts had moved slightly within the oversized holes in the shroud support cone during plant operation. Although the holes are round when viewed perpendicularly from the shroud support cone, their projection on a horizontal plane is oval-shaped due to the angle of the support cone. Thus, the toggle bolts are allowed to move up the inclined surface of the shroud support.

NMPC calculated the maximum tie rod looseness that could have been caused by movement of the toggle bolts within the holes in the shroud support cones. The maximum upward movement (as measured by turning the tie rod nut on the 90° tie rod) was 0.151 inch; whereas, the differential thermal expansion in the tie rod is 0.155 inch, which is greater than the looseness at the shroud support cone. Therefore, although the initial installation mechanical preload was lost, some thermal preload would have remained. Significantly more thermal preload remained at the other three tie rod locations where the measured upward movement ranged from 0.054 inches to 0.093 inches.

Visual inspection revealed that the lower support wedge latch at the 90° location was broken. A piece of the latch that was missing was later found on the lower support cone at approximately azimuth 330°. After examining photographs and video tapes, NMPC stated that fatigue did not appear to be the failure mechanism. There was no visible evidence of plastic deformation, which would be necessary for a single-overload type of failure. The failure surface appeared to be consistent with a stress corrosion failure under high stress. Results from a metallurgical evaluation at a GE testing laboratory are expected to be available by May 20, 1997. Pending completion of those examinations, NMPC believes that the most likely failure mechanism is stress corrosion, although failure from overload has not been ruled out. As the NRC staff understands this information, this conclusion appears reasonable.

Videotape inspection of the other three lower wedge latches indicated that they are all in one piece, but the latch at the 350° location appeared to be bent. In addition, the lower spring wedges showed minor evidence of local hard contact with the latches. Since the latch material, alloy X-750, is harder than the lower spring wedge material, Type 316 low-carbon stainless steel, the lower spring wedges are likely to show surface wear before the latches.

A similar latch is used in each mid-support assembly, and two similar latches are used in each upper support assembly. NMPC observed the latches on the middle and upper supports, and all 12 appear to be in good condition. Because of design differences, these latches cannot be loaded as severely as the lower wedge latches. The contact force between the RPV and the shroud repair is much smaller at these locations than the contact force at the lower wedge. In

addition, these latches are not loaded during plant heat-up. The lower wedge at the 90° location had dropped to the bottom of the post on the lower spring. The lower wedge at the 350° location appeared to be approximately 1/8 inch below its normal position. The other two wedges were at their normal positions.

Contact sliding marks have been observed on the RPV wall at the 166° location above the lower and upper contact points of the upper spring assembly. Sliding marks are not evident at any other contact points for the upper springs located at the 90°, 270°, and 350° locations. Visual inspection of all mid-support contacts confirmed that there was contact with the RPV surface in the cold condition. During normal operation, the mid-support compression on the RPV increases because of thermal conditions. Thus, the functions of providing a load path from the tie rod to the RPV and increasing the natural frequency of the tie rod assemblies were maintained. During inspections and tests related to the root cause investigation of the lower support wedge latch failures, two mid-supports at azimuths 90° and 166° had to be repositioned. New mid-supports have been fabricated and installed with the original design preload at these locations. The mid-supports at the 270° and 350° locations have also been verified for proper preload. Thus, the required support configuration will be maintained during future operation.

The root cause for the tie rod degradation affecting both the tie rod nut and the lower latch, is attributed to the movement of the toggle bolts within oversized lower support bolt holes. The installation procedures did not contain specific criteria for locating the toggle bolts during installation of the lower support. The lower support toggle bolts are nominally 4.000 inches in diameter. The measured electric discharge machining (EDM) holes in the shroud cone ranged from 4.090 inches to 4.110 inches. Since the position of the lower support within the machined holes was not procedurally controlled during installation, the relative position of the bolts within the holes was randomly located.

During heat-up, the expansion of the shroud and tie rods generated a force sufficient to overcome the installation friction forces and slightly move the lower support toggle bolt assembly up the shroud cone. This translated into a vertical movement of the tie rod, loading the latch on the lower spring wedge and causing its failure. These latches were not designed to accommodate differential movement between the RPV wall and the lower spring wedge during normal and transient conditions in the event the lower wedge got stuck at the RPV wall. This also caused the loss in preload on the tie rod nut, described earlier. The NRC staff has reviewed NMPC's analysis of the tie rod degradation and agrees with NMPC's conclusions.

In the original design of the lower wedge, it was assumed that the wedges would slide on the RPV wall and accommodate differential thermal expansion between the tie rod assembly and the RPV. However, the actual frictional force between the wedge and the RPV was higher than anticipated and sufficient to prevent movement of the wedges at the RPV interface during thermal growth of the tie rod assembly. This caused the retainer clip (or latch) to stretch

between its attachment points on the tie rod and the wedge. A portion of the retainer clip was overstressed and failed because of stress corrosion cracking.

On the basis of its review (as discussed above), the NRC staff believes NMPC's preliminary root cause determination is reasonable. However, even if the failure resulted from overstress, it appears that the modification would address the failure.

5.2.2 Modification of the Lower Support Latch Design

The lower support latch (or retainer clip) has been redesigned to accommodate movement during normal and transient conditions. The redesigned latches were installed before core reload. The new design is substantially more flexible and takes into account the various potential sliding cases. These cases include combinations of sliding at the RPV wall/lower wedge and lower wedge/lower spring interfaces. The largest latch displacement is postulated during the initial heat-up and hydrotest where the wedge is projected to slide at the spring interface but remain locked at the RPV interface. During a potential cooldown to a ambient conditions the wedge is projected to slide only at the RPV interface. Finally, during a heat-up to full power, the wedge is assumed to slide at the spring interface with no movement at the RPV surface. The calculated latch displacement for this worst-case scenario has been determined to be 0.182 inches. The calculated stress in the new latch for this displacement has been determined to be 60 percent of the allowable stress. In addition, the latches have been evaluated for potential damage due to SCC. Utilizing a Stress Rule Index (SRI) methodology, NMPC has demonstrated that for the worst-case sliding, stress corrosion will not occur in the next operating cycle. However, if a most probable wedge/RPV/spring interface sliding scenario is assumed, then no corrosion is projected for the remaining life of the plant.

5.2.3 NRC Staff Evaluation of the Structural Analysis of the Redesigned Lower Support Latch

On the basis of its review of the new retainer clip design, the NRC staff raised several concerns. One concern was that the slightly sloping surface on the lower wedge would preferentially cause the wedge to slide in one direction and get stuck in the other direction. NMPC, in response, clarified that the purpose of the angle on the lower wedge is to facilitate moving the lower wedge into position and creating the contact force with the RPV wall. This slight slope on the wedge does not influence whether sliding occurs at the wedge/spring interface or the RPV wall/wedge interface. Therefore, during plant operation, sliding is expected to occur at whichever surface has the lowest friction factor. As discussed in GENE Report B13-01739-22, sliding is likely to occur at the wedge/spring interface because the machined surfaces and dissimilar materials will result in a low friction factor.

In response to the NRC staff's inquiry about the feasibility of lowering the lateral load at the wedges to facilitate sliding, NMPC discussed the considerations in determining the lateral design loads on the wedges. The intent of the lower spring design was to ensure that positive contact exists

during shutdown conditions, and the 0.010-inch radial contact at the lower spring represents a minimal condition that meets this requirement. Recognizing that the measurements for matching the lower wedge have to be taken remotely from the reactor refueling floor, the 0.010-inch interference represents a value that confidently ensures contact at the lower spring support. Without radial support at the lower wedge, the only potential problem is vibration. However, as demonstrated at the 90° tie rod location during the last operating cycle (which lost contact), and by analysis, NMPC concluded that vibration of the tie rod assembly is not an issue. NMPC's responses as discussed above, reasonably address the NRC staff's concerns.

The maximum possible differential vertical displacement of the lower wedges and the probable wedge movement has been determined. Since the potential exists that the wedges may not always slide at the spring interfaces due to unanticipated forces on the wedge during various operational transients, the design of the latches has been based upon the maximum estimated vertical displacement. The maximum displacement assumed for the latch design exceeds the maximum vertical displacement of the tie rod hardware at the lower spring elevation as shown in Table 4.1 of GENE Report B12-01739-22.

During certain plant operating and test conditions (e.g., during a hydrotest), the radial contact force on the wedges is likely to be minimal. Under such conditions, the wedges could potentially slide circumferentially along the RPV surface and remain stuck in that position. In response to NRC staff concerns that this could impose an additional torsional moment on the latch during subsequent plant operation, NMPC indicated that these considerations have been factored into the new latch design. The ratio of the width of the wedge to the potential gap between the wedge and the RPV wall is very large. Therefore, potential angularity of the wedge in a loss-of-contact event is extremely small and any applied force on the wedge will cause it to readjust the orientation of the wedge to distribute the contact forces across the entire face of the wedge surface, and not allow an edge to become stuck. Even if it is assumed that the wedge would become stuck on one edge, the latch mechanism has lateral clearance with the wedge and the spring interface slots, so that the latch will reposition itself without any torsional loads being applied.

The new latch design stresses have been calculated to be substantially lower than the original latch design stresses during various postulated operating, transient, and accident conditions. The membrane plus bending stresses in the new latch design are lower by a factor of 8.6 in comparison with the original latch stresses. The corresponding reduction factor for the membrane, bending, and peak stresses is 12.8. This indicates a substantial improvement in the design margin for the new latches.

5.2.4 NRC Staff Analysis of Shroud and Tie Rod Assembly

Section 4 of GE Report GENE B13-01722-40, "Shroud Repair Anomalies--Nine Mile Point Unit 1, RF014," dated April 1997, contains an evaluation of the safe operation of the shroud and the tie rod assembly with the existing ligaments in the horizontal and vertical welds. NMPC's assessment indicates that the safe operation of NMP1 was not impaired with the as-found condition of the tie

rod assembly and an assumption that all horizontal welds were 360° through-wall cracked. It is known, through actual inspection of the shroud, that in the regions where inspection has been performed, no through-wall cracks or 360° crack lengths had been found in any of the horizontal welds. Therefore, NMPC's evaluation is considered conservative.

NMPC has analyzed the as-found condition of the shroud vertical welds and has established that the plant can be operated safely. A conservative interval for reinspection of the welds has been established. NMPC has committed to reinspect the tie rods, which will include checking the latch devices and checking the tie rod nuts for tightness, after approximately 10,600 hours of operation. On the basis of its review of the structural integrity of the re-designed latch and the core shroud vertical welds discussed earlier, the NRC staff finds the inspection interval acceptable.

5.2.5 Revised Installation Procedures for Tie Rod Assemblies

As described in NMPC's Document NMP-SHD-003, Revision 1, "Lower Wedge Latch Replacement and Tie Rod Torque Checks," including Special Process Control Sheet SPCS#01, Revision 1, NMPC has developed improved installation procedures to ensure that the tie rod assemblies are installed with no gaps at the upper end of the support cone holes. The improved procedure requires that each tie rod be jacked at three locations during tie rod nut torquing to remove any gaps associated with installation tolerances. Jacks are placed under the lower support, on the RPV side of the lower support, to push it up the shroud cone. This removes the clearances between the toggle bolts and the shroud side of the cone holes.

In its letter to the NRC dated April 27, 1997, "Responses to NRC Staff Questions Provided During Telephone Conversations of April 10, 11, and 24, 1997, on Core Shroud Cracks and Repair," NMPC states that there are no plant operating, transient, or test conditions that will cause the toggle bolts to slide down the shroud support cone. The temperature of the shroud/RPV annulus would have to be higher than the RPV and shroud temperatures, in addition to a complete loss of preload in the tie rods to cause such a movement. This combination of events does not occur during operation.

After installation, and in accordance with the revised procedures, inspections were completed on each tie rod assembly to verify proper contact by verifying the absence of gaps. These inspections revealed that the middle support was no longer in contact with the RPV on the 90° and 166° tie rod. This resulted from the movement of the lower support assembly up the cone toward the shroud. The middle support dimensions were obtained and new middle supports were installed before reload. Other locations on the tie rod assemblies with the potential for gaps and non-conforming conditions were inspected. No additional deficiencies were noted.

6.0 FUTURE INSPECTIONS

In its letter of April 30, 1997, NMPC also stated that it will propose an inspection plan for the next scheduled outage and submit the plan to the NRC at least 3 months before the outage is scheduled to begin. This plan should

provide details regarding the inspection of the shroud repair components; the shroud repair anchorages; and the shroud's horizontal, vertical, and ring segment welds. The plan will specify the inspection methods to be used, including the provisions for sample expansion.

7.0 CONCLUSION

7.1 Inspection

The NRC staff has determined that the scope and methods used for the inspections of the NMP1 core shroud during the spring 1997 refueling outage met or exceeded the requirements of BWRVIP-07, and are, therefore, acceptable.

7.2 Structural Integrity

The NRC staff has concluded that results of the analyses show that the core shroud maintained the required ASME Code margins during the last operating cycle. The review considered the loss of pre-load for the tie-rod repairs. Additional analyses submitted by the licensee pursuant to 10 CFR 50.55a implemented by ASME Code Section XI for the vertical welds have been shown to have sufficient ligaments to support the tie-rod repair for at least 10,600 additional hours of operation. The staff performed an independent analysis and determined that the licensee's analyses appear reasonable.

7.3 Tie Rod Repair

On the basis of its review of the root cause analysis of the shroud repair anomalies, the NRC staff concludes that the tie rod was loosened by the movement of the toggle bolts within oversized bolt holes. NMPC's new installation procedures include measures to prevent tie rod looseness and maintain tie rod vertical forces as intended in the original design. The root cause of the latch failure was larger-than-anticipated vertical displacements of the latch, which overstressed the latch and most likely subjected it to SCC. The new latch has been redesigned to accommodate larger vertical displacements while maintaining its original function of locking the wedge to the lower spring structure. On the basis of a most probable sliding scenario at the wedge/RPV interface, no failure is projected for the remaining life of NMP1. The calculated stresses are within the ASME Code allowable values and the latch has been analyzed to be resistant to stress corrosion for a minimum of 2 years assuming worst-case displacements in the latch. NMPC's root cause evaluation and corrective actions offer reasonable assurance that the tie rods will perform their intended function. The root cause evaluation will be confirmed in the near term by ongoing evaluations. Future inspections will monitor the conditions of the tie rod assemblies.

The NRC staff has determined that the modified tie-rod repair is an acceptable alternative to a repair in accordance with ASME Code Section XI, pursuant to 10 CFR 50.55a(a)(3)(i). The alternative provides an acceptable level of quality and safety because the ASME Code margins of safety will be maintained for the operating period. This NRC approval under 10 CFR 50.55a(a)(3)(i) is contingent upon NMPC (1) maintaining reactor coolant chemistry within the

guidelines set forth in Electric Power Research Institute technical report TR-103515, "BWR Water Chemistry Guidelines-1996 Revision," in accordance with its commitment by letter dated April 30, 1997, and (2) submitting, within 60 days, an application for a license amendment that addresses this matter in accordance with its commitment by letter dated May 7, 1997. Failure to satisfy either of these conditions will render this approval null and void.

8.0 BIBLIOGRAPHY

1. Letter from L.B. Marsh, U.S. NRC to Niagara Mohawk Power Corporation dated March 31, 1995, and attached Safety Evaluation by the Office of Nuclear Reactor Regulation relating to the Core Shroud Repair at Nine Mile Point Unit 1.

2. Letter from Niagara Mohawk Power Corporation to U.S. NRC (NMPIL 0927), dated March 25, 1995.

3. Letter from Niagara Mohawk Power Corporation to U.S. NRC (NMPIL 1067), dated April 30, 1995.

4. Letter from Niagara Mohawk Power Corporation to U.S. NRC (NMPIL 1111), dated August 14, 1996, with enclosure providing final design documentation for modifications to the core shroud repair assemblies installed in March 1995.

5. Letter from NRC to Niagara Mohawk Power Corporation, dated March 3 1997, with enclosed Safety Evaluation by the Office of Nuclear Reactor Regulation relating to the Modifications for Correcting Deviations identified in Core Shroud Repairs at Nine Mile Point Unit 1 (TAC No. M90102).

6. Letter from Niagara Mohawk Corporation to NRC, dated January 6, 1995, with enclosed design report of Core Shroud repair hardware. Supplemented by letters dated January 23, 26, February 14, 24, 28, March 7, 9, 13 (2 letters), 14 (2 letters), 23, 27, 28 and 30 1995.

7. Letter from Niagara Mohawk Corporation to NRC, NMPIL 1200 dated April 8, 1997, (Enclosure 2) GENE B13-01739-40 "Shroud Repair Anomalies Nine Mile Point 1, RF014," April 1997.

8. GENE-B13-01739-22, Revision 0, Nine Mile Point 1, "Design Report for Improved Shroud Repair Lower Support Latches," April 1997.

9. Niagara Mohawk Document NMP-SHD-003, Revision 1, "Lower Wedge Latch Replacement and Tie Rod Torque Checks," including Special Process Control Sheet SPCS#01, Revision 1.

10. Safety Evaluation of Niagara Mohawk Power Corporation (NMPC) Reinspection Plans for Nine Mile Point Unit 1 Core Shroud, NRC, March 3, 1997.

11. "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03)," EPRI Report TR-105696, October 1995.

12. "BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)," EPRI Report TR-105747, February 1996.

13. "Reactor Pressure Vessel and Internals Examinations Guidelines (BWRVIP-03)," EPRI TR-105696, October 1995.

14. "Assessment of the Vertical Weld Cracking on the NMP1 Shroud," Report No. GE-NE-523-B13-01869-043, Rev. 0, April 1997. Prepared for Niagara Mohawk Power Company by GE Nuclear Energy.

15. "Shroud Repair Anomalies, Nine Mile Point Unit 1, RF014," Report No. B13-01739-40, Rev. 0, Class III, April 1997, Prepared for Niagara Mohawk Power Company by GE Nuclear Energy.

16. "Nine Mile Point Unit 1 Core Shroud Cracking Evaluation," Prepared for George Inch, Niagara Mohawk Power Company, by Richard E. Smith, Altran Corp., April 3, 1997.

17. "Analysis of Nine Mile Point Unit 1 Shroud Welds V9 and Weld V10 Cracking," April 1997, Report No. MPM-497439, MPM Technologies.

18. Safety Evaluation 96-018, pursuant to 10 CFR 50.59 Rev. 1.

19. "Concerns Regarding Operation of Nine Mile Point 1," letter dated April 9, 1997, from D. Lochbaum, Union of Concerned Scientists to S. Bajwa, NRC.

20. "Potential Unanalyzed Operation of Nine Mile Point Unit 1 Core Shroud Vertical Cracks," letter dated April 17, 1997, from D. Lochbaum, Union of Concerned Scientists to S. Bajwa, NRC.

21. "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," GENE-523-113-0894, Rev. 1, General Electric Company, March 1995. Prepared for the BWRVIP Assessment Committee.

22. "BWR Water Chemistry Guidelines, 1993 Revision, Normal and Hydrogen Water Chemistry," EPRI TR-103515, February 1994.

23. "Response to April 17, 1997, letter from Union of Concerned Scientists to the NRC," letter dated April 25, 1997, from NMPC to NRC.

24. "Response to Union of Concerned Scientists letter dated April 9, 1997, Regarding Nine Mile Point Unit 1," letter dated April 26, 1997, from NMPC to NRC.

25. "Responses to NRC Staff Questions Provided during Telephone Conversations of April 10, 11, and 24, 1997, on Core Shroud Cracks and Repair," letter dated April 27, 1997, from NMPC to NRC.

26. "Responses to NRC Staff Questions Provided during Telephone Conversations of April 22 1997, on Core Shroud Cracks and Repair," letter dated April 27, 1997, from NMPC to NRC.

27. Brunswick Steam Electric Plant No. 1, Docket No. 50-325/License No. DPR-71 NRC Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," letter dated June 21, 1995, from R. Anderson of Carolina Power and Light Co. to the NRC.

28. Letter from Niagara Mohawk Power Corporation to U.S. NRC (NMPIL 1217), dated May 7, 1995.

Dated: May 8, 1997

ENCLOSURE

Figures, tables, and Appendix C from General Electric Nuclear Energy Document GE-NE-B13-01869-043, Revision 0.

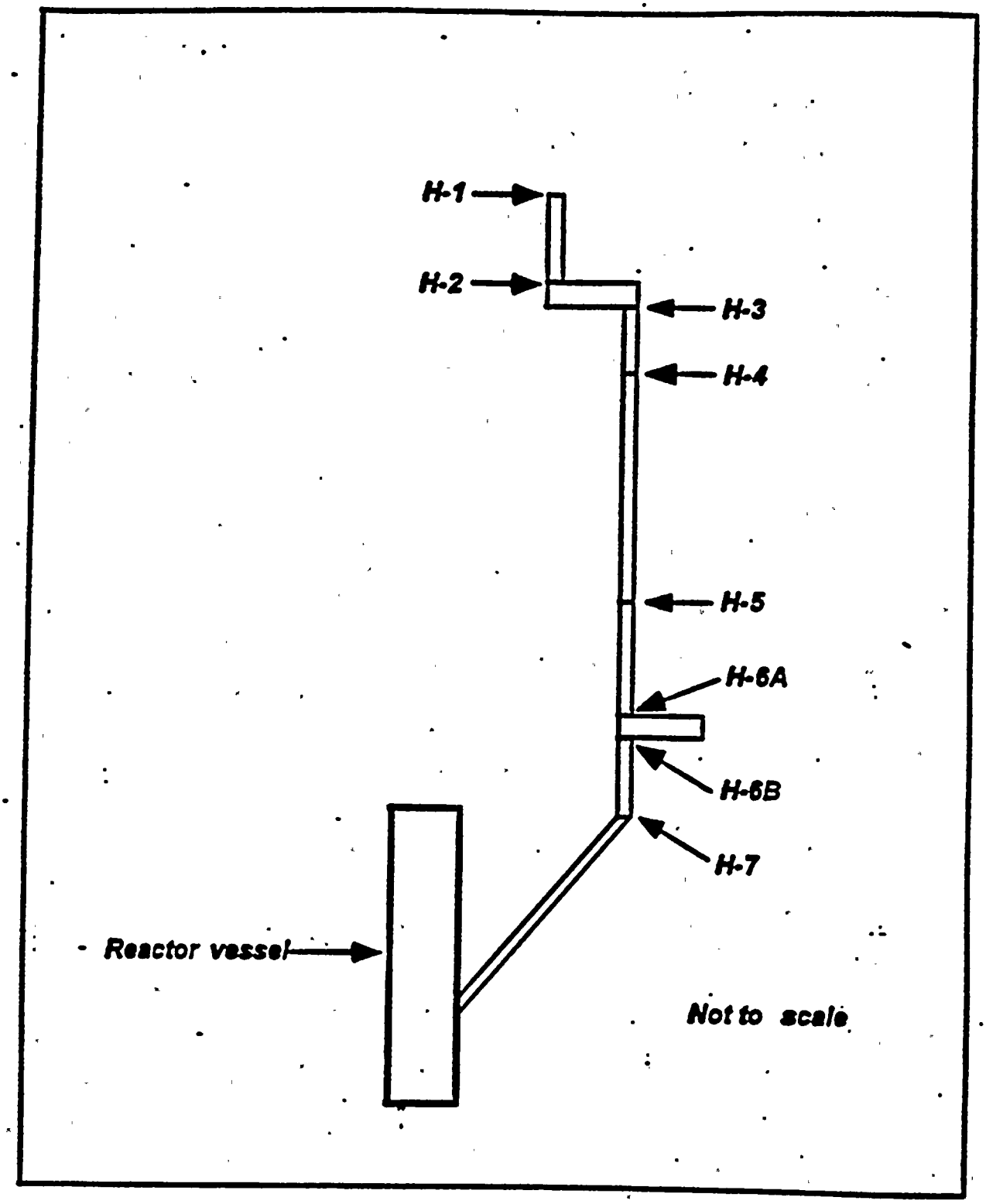

Figure 1-1 NMP1 Shroud Weld Locations, Cross Sectional View

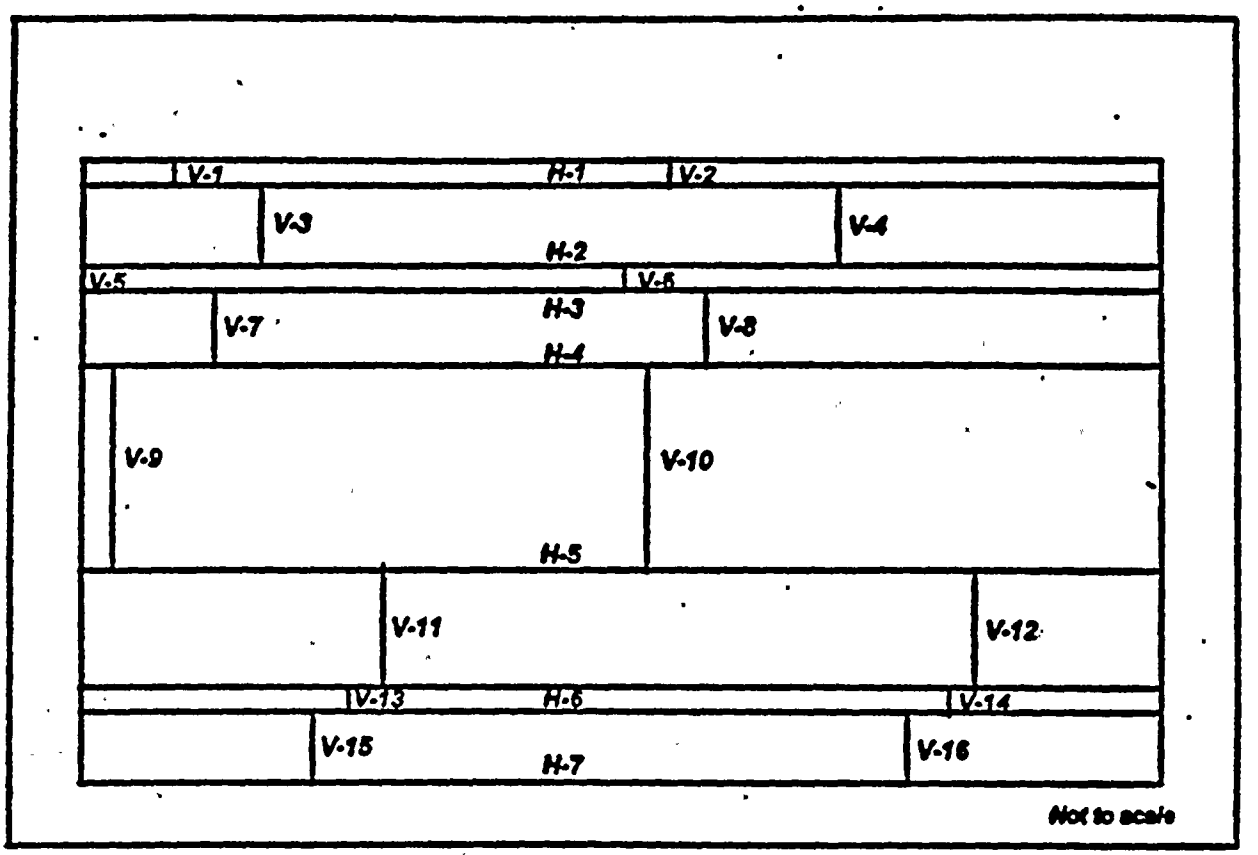

Figure 1-2 NMP1 Shroud Weld Locations

Table 2-2
Summary of Recent Shroud Vertical Weld Inspections (RFO 14)

Weld	Weld Length (in)	Inspection Coverage*	Shroud ID/OD	Exam Type	Flaw Length
V-3	31.25	15" Left 15" Right	OD	UT	1.5" ID, Right HAZ 0.8" OD, Right HAZ
V-4	31.25	22" Left 11" Right	OD	UT	22"ID Left HAZ, 1.5"ID Right HAZ
V-5 ring		Not located	NA	NA	NA
V-6 ring		Not located	NA	NA	NA
V-7	18.5	9" Left 11" Right	OD	UT	No Indications
V-8	18.5	5.5" Left 9.5" Right	OD	UT	No Indications
V-9 shell	90.12	100%	ID and OD	EVT-1	Indications on over 90% OD right HAZ Minor cracking on OD left side and on ID both sides
		80"	OD	UT	Numerous indications on OD, Left HAZ Two minor flaws on ID, Right HAZ
V-10	90.12	100%	ID and OD	EVT-1	Cracking on OD, Right HAZ Cracking on ID, Left and Right HAZ
		84"	OD	UT	Flaws detected on > 80% on OD, Right HAZ Flaws detected on > 10% on OD, Left HAZ
V-11	63.5	100% OD 50% ID	ID and OD	EVT-1	No Indications
V-12	63.5	100% OD 50% ID	ID and OD	EVT-1	6" OD, Right HAZ
V-15	22.13	11" Left 11" Right	OD	UT	6" ID, Left HAZ 2.2" ID, Right HAZ
V-16	22.13	100%	OD	EVT-1	.75" OD, Left HAZ
		10.5" Left 20" Right	OD	UT	5" ID Left HAZ 4" ID Right HAZ 3" ID Left HAZ from right side exam

* The inspected regions indicated on each side of the weld are not necessarily coincident, hence the integrated inspection coverage may be less than indicated, but has been taken into account in determining the uncracked ligament length.

Appendix C
Shroud Inspection Summary

The following is a weld by weld summary detailing the scope of inspections and results of the shroud examinations performed to date.

Weld V-3

Performed ultrasonic examination of approximately 15 inches of each side of the weld from the shroud OD surface. Approximately 1.5" of flaw was detected on the ID surface and 0.8" of flaw on the OD surface.

Weld V-4

Performed ultrasonic examination of approximately 11" of the left HAZ and 22" of the right HAZ. ID flaws were detected over the entire examined length of the left HAZ and 1.5" of flaw was detected on the ID of the right HAZ.

Weld V-7

Performed ultrasonic examination of approximately 9" of the left HAZ and 11" of the right HAZ. No flaws were detected during the examination..

Weld V-8

Performed ultrasonic examination of approximately 5.5" of the left HAZ and 9.5" of the right HAZ. No flaws were detected during the examination.

Weld V-9

Performed ultrasonic examination from the shroud OD surface for approximately the entire length of both the left and right HAZs as well as EVT from both the ID and the OD. Visual cracking was detected over greater than 90% of the right HAZ on the OD and minimal cracking was detected on the ID in both the left an right HAZs. Minor cracking was also detected on the OD in the left HAZ. The cracks detected visually on the shroud ID surface were found to be predominantly transverse to the weld whereas the cracking detected visually on the shroud OD surface was mostly parallel to the weld with components that branched transverse to the weld. Ultrasonic examinations of essentially the entire length of the weld was performed from the shroud OD surface and detected numerous flaws over the length of the left HAZ emanating from the shroud OD surface. Two small flaws on the ID surface were detected in the right HAZ.

Weld V-10

Performed ultrasonic examination from the shroud OD surface for approximately the entire length of both the left and right HAZs as well as EVT from both the ID and the OD. Flaws were detected on greater than 80% of the right HAZ on the OD surface and greater than 50% of the left HAZ revealed flaws on the OD surface. The EVT examination revealed cracking in the left and right

HAZs on the OD surface for most of the length of the weld and on the ID in both the left and right HAZs. The cracks detected visually on the shroud ID surface were found to be predominantly transverse to the weld whereas the cracking detected visually on the shroud OD surface was mostly parallel to the weld with components that branched transverse to the weld.

Weld V-11

EVT examinations were performed on the accessible weld length from both the ID and the OD of both the left and right HAZs. No cracking was detected during the examination.

Weld V-12

EVT examinations were performed on the accessible weld length from both the ID and the OD of both the left and right HAZs. One 6" crack was detected on the length OD surface in the right HAZ. No other cracking was detected.

Weld V-15

Ultrasonic examination was performed from the shroud OD surface on approximately 11 inches of both the left and right HAZs. One 6" flaw was detected in the left HAZ on the ID surface and several ID flaws totaling 2.2" in total length was detected on the ID in the right HAZ. No flaw detected in either HAZ was greater than 10% through wall.

Weld V-16

Ultrasonic examination was perfotmed from the shroud OD surface of approximately 10.5" of left HAZ. Two flaws were detected on the ID surface. One flaw was 5" in length, 10% through wall. The other ID flaw in the left HAZ was detected from the scan on the right HAZ and was 3" long and 30% through wall. Approximately 22 inches of the right HAZ was examine from the shroud OD surface. One flaw was detected on the ID which measured 4" in length and 21% through wall. An EVT examination of both HAZs from the shroud OD surface revealed one crack in the left HAZ.

Recent Inspection Results for Shroud Horizontal Welds

In addition to the shroud vertical weld inspections, the horizontal welds H-2, H-4, H-5, H-6a, H-6b, and H-7 were also inspected for analytical purposes, to evaluate the overall integrity of the shroud using assumptions of worst case cracking of the vertical welds.

Weld H-2

Ultrasonic examination was performed from the shroud OD surface of approximately 24 inches of the upper HAZ adjacent to weld V-4. Approximately 7 inches of intermittent flaws were detected on the OD surface, with the deepest area having a through wall depth of .22 inches.

Weld H-4

Ultrasonic examination from the shroud OD surface was performed on approximately 60% of the lower HAZ. ID and/or OD flaws were detected intermittently throughout the examination area. Some ID flaws were detected in the upper HAZ. Approximately 32 inches of the upper HAZ was ultrasonically examined. 3 inches of shallow OD flaws were detected in the upper HAZ and one 6 inch long ID flaw was detected with the maximum through wall depth of .23 inches. . An EVT examination of the OD was performed of over 70% of the upper and lower HAZs. Cracks were detected in both the upper and lower HAZs.

Weld H-5

Ultrasonic examination from the shroud OD surface was performed on approximately 30% of the upper and lower HAZs. OD and ID flaws were detected in the upper HAZ only. No flaws were detected in the lower HAZ. EVT of approximately 60% of the shroud OD surface revealed cracks intermittently in both the upper and lower HAZs. Most of the flaws detected visually on the OD surface were oriented perpendicular to the weld. No flaws were detected in the upper HAZ at the intersections of welds V9 or V10.

Weld H-6A

Ultrasonic examination was performed on both the upper and lower HAZs of approximately 30% of the circumference from the shroud OD surface. Flaws were detected on the OD surface of the lower HAZ only. No flaws were detected in the upper HAZ or on the ID of either HAZ.

Weld H-6B

Ultrasonic examination was performed on both the upper and lower HAZs of approximately 30% of the circumference from the shroud OD surface. Flaws were detected on the OD surface of the upper HAZ only. No flaws were detected in the lower HAZ or on the ID of either HAZ.

Weld H-7

Ultrasonic examination was performed for the shroud OD surface on the upper HAZ on approximately 30% of the circumference. No flaws were detected during the examination.

Weld H-8

Ultrasonic examination was performed for the shroud OD surface on the lower HAZ on approximately 30% of the circumference. A flaw which was identified by UT during a prior

outage was located as well as one additional flaw in the same area. This flaw was ultrasonically sized to be of lesser through wall depth than in RFO13. A review of the previous data indicates that the previous sizing performed was very conservative. An EVT was performed on approximately 30% of the circumference from the shroud OD surface. Of the five small cracks visually detected during RFO13 only 1 was visible during this inspection. The inspection in the area of the other four was hampered by the placement of a Tie Rod support which prevented a good EVT inspection. Cracks were visually detected in three new locations in the upper HAZ. The largest of these cracks (9"-12") is located predominantly in the ring segment Upper HAZ and runs into the weld toe and back into the ring segment.

Weld H-9

An EVT examination was performed in one area 26 inches long. No indications were noted during the examination.

Table 5-2
Allowable Flaw Sizes for the Nine Mile Point Unit 1 Shroud Vertical Welds

(1) Weld ID	(2) Weld Length, in	(3) Allowable Through wall crack length, in. LEFM	Limit Load	(4) Minimum required ligament, in.	(5) Min. Ligament including crack growth (two years) and Inspection Uncertainty, in. (Note 1)	(6) Available Equivalent Uncracked Ligament Length, in.
V-3, V-4	31.25	-	29.97	1.28	3.63	7.3 (V-3) Note 2 (V-4)
V-7, V-8	18.50	18.3	17.78	0.72	3.07	9.0 (V-7) 5.6 (V-8)
V-9, V-10	90.12	75.40	86.61	14.72	17.07	Note 2 (V-9) Note 2 (V-10)
V-11, V-12	63.50	58.20	61.03	5.30	9.30 Note 3	31.75 (V-11) 25.75 (V-12)
V-15, V-16	22.13	-	19.53	2.46	4.81	Note 4 (V-15) 5.5 (V-16)

Notes

1. Based on crack growth of 1.6 in. and UT inspection uncertainty of 2 x 0.375 inch at each crack tip for length sizing.
2. Meets requirements based on further evaluation reported in Subsection 5.3.
3. The minimum ligament for EVT inspection is larger to account for greater uncertainty in the visual inspection. The uncertainty factor applied is equal to 2 x 1.2 in.
4. The equivalent length after subtracting crack growth and inspection uncertainty is 2.89 in. which is greater than the required ligament of 2.46 in. and thus acceptable.