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Withholding Granted 5/5/97

SUBJECT: Forwards non-proprietary & proprietary responses to 970410 & 11 telcons RAI re core shroud weld cracking & tie rod repair.Proprietary response withheld,per 10CFR2.790(b)(1).

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MARTIN J. McCORMICK JR. P.E.
Vice President
Nuclear Engineering

April 23, 1997
NMP1L 1206

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

RE: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

*Subject:* ***Responses to NRC Staff Questions Provided During Telephone Conversations of April 10 and 11, 1997, on Core Shroud Cracks and Repair***

*Reference:* ***Letter, Darl S. Hood (NRC) to Niagara Mohawk Power Corporation, Dated April 18, 1997***

Gentlemen:

By letter dated April 8, 1997, Niagara Mohawk Power Corporation (NMPC) submitted to the NRC the results of the inspection of Nine Mile Point Unit 1 (NMP1) Core Shroud and Stabilizer Assemblies conducted during the Spring 1997 refueling outage. The submittal of April 8, 1997 also requested approval for a proposed modification to the stabilizer assemblies as allowed by 10CFR50.55a.

Following the submittal of April 8, 1997, the staff requested additional information from NMPC pertaining to the core shroud inspection results and the proposed stabilizer assembly modifications (during telephone conversations on April 10 and 11, 1997). The staff also requested that NMPC respond in writing to this request for additional information following the public meeting on April 14, 1997. Except for the responses to questions 1 and 2 of Enclosure 1, the responses to the request for additional information are provided in Enclosures 1, 2, and 3. The responses to questions 1 and 2 of Enclosure 1 will be submitted by NMPC following independent verification of the results of the ultrasonic testing. The expected submittal date is April 25, 1997.

NMPC has also removed two boat shaped samples from the NMP1 core shroud. NMPC will submit a report of the results of the analysis of the samples on or before September 30, 1997.

Enclosure 1 of this submittal addresses the questions presented in Enclosure 2 of the Reference. Enclosure 2 of this submittal addresses the questions presented in Enclosure 3 of the Reference. Enclosure 2 is considered by its preparer, General Electric (GE), to contain proprietary information exempt from disclosure pursuant to 10CFR2.790. Therefore, on behalf of GE, NMPC hereby makes application to withhold this document from public disclosure in accordance with 10CFR2.790 (b)(1). A non-proprietary version of these documents has been included with this letter as Enclosure 3. An affidavit executed by GE detailing the reasons for the request to withhold the proprietary information has been included in Enclosure 4.

Sincerely,

M. J. McCormick Jr.

Martin J. McCormick Jr.
Vice President - Nuclear Engineering

MJM/MSL/lmc
Enclosures

xc: Mr. H. J. Miller, Regional Administrator, Region I
Mr. B. S. Norris, Senior Resident Inspector
Mr. S. S. Bajwa, Acting Director, Project Directorate I-1, NRR
Mr. D. S. Hood, Senior Project Manager, NRR
Records Management

# ENCLOSURE 4

# GENERAL ELECTRIC AFFIDAVIT

## General Electric Company

## AFFIDAVIT

I, **George B. Stramback**, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the attachments to GE letter Gerald A. Deaver to Steve Leonard, *Responses to NRC Staff Questions on the Shroud Repair* (GE Proprietary Information), dated April 21, 1997. The proprietary information is delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains GE response information about the detailed results of analytical models, methods and processes, including computer codes, which GE has developed and applied to perform evaluations of indications in the core shroud for the BWR.

The development and approval of the BWR Shroud Repair Program was achieved at a significant cost, on the order of one million dollars, to GE.

The development of the evaluation process contained in the paragraph (2) document along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA )
) ss:
COUNTY OF SANTA CLARA )

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 21st day of April 1997.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 21st day of April 1997.

Notary Public, State of California

JULIE A. CURTIS
Commission # 1113834
Notary Public — California
Santa Clara County
My Comm. Expires Oct 20, 2000

# ENCLOSURE 1

# REQUEST FOR ADDITIONAL INFORMATION REGARDING CORE SHROUD WELD CRACKING AND TIE ROD REPAIR

(Reference Letter, Darl S. Hood (NRC) to Niagara Mohawk Power Corporation, Dated April 18, 1997)

## ENCLOSURE 1

## REQUEST FOR ADDITIONAL INFORMATION REGARDING CORE SHROUD WELD CRACKING AND TIE ROD REPAIR NINE MILE POINT NUCLEAR STATION UNIT 1

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*Request for Information #1*

*Please discuss how the uncertainty factors used in crack growth calculations (UT and VT measurements of flaw length and depth) for the vertical welds were determined and also address how the BWRVIP guidelines on inspection uncertainties were met.*

Required Response #1

Required Response #1 will be submitted following independent verification of the results of the ultrasonic testing.

*Request for Information #2*

*It appears that the indication maps for vertical welds V10 and V9 are not consistent with the corresponding flaw data plots. In the flaw data plots, the uncracked areas are much larger than that shown on the corresponding indication maps. Please explain the discrepancy and discuss its impact on the determination of reinspection interval.*

Required Response #2

The Enclosure 1 analysis was performed based on UT data which had not been accepted by the NMPC level 3 NDE inspectors. The Appendix C UT data was the latest UT data which was available at the time of the submittal. The NMPC submittal letter noted that the final NDE data was not complete. The Enclosure 1 analysis Figure 5-7 did bound the data in Appendix C, however, the idealized histogram of the V9 UT data was not based on the UT data in Appendix C.

Required Response #2 will be submitted following independent verification of the results of the ultrasonic testing.

*Request for Information #3*

*In Figure 5-6, only the crack growth in the wall thickness (depth) direction was considered. Please provide justification for not considering the crack growth in the axial direction (parallel to vertical weld) and the potential crack initiation and growth in the uncracked areas. If we assume that cracks will initiate and grow in the uncracked areas, what would be the acceptable*

*reinspection interval for vertical weld V9? The bounding case would be assuming the uncracked areas to be cracked with zero depth.*

Required Response #3

Crack growth, both in the length and depth direction, was considered in the evaluation of the limiting crack. The evaluation also considered potential crack initiation and growth in the uncracked areas. Therefore, the reinspection interval of 10,600 hours includes consideration of the potential for axial crack growth including uncracked areas.

*Request for Information #4*

*Limited inspection was performed on horizontal welds H8 (30% of circumference) and H9 (26 inches in circumference in one area). Some minor cracking was found in weld H8. Please provide your basis that the extent of the weld inspection was adequate and that weld will retain its integrity at the end of the proposed period of operation assuring the core will remain adequately supported.*

Required Response #4

The inspection plan for the shroud submitted to the staff on February 7, 1997, described the re-inspection plans for the H8 weld. These plans discussed that an enhanced visual examination of the H8 weld would be performed in the regions for which the Ultrasonic Test (UT) examination was performed in 1995, and that the previously identified visual indications on the shroud support ring Heat Affected Zone (HAZ) of the H8 weld would be re-inspected. This re-inspection plan was based on the 1995 UT inspection coverage and analysis which determined that the required UT inspection interval for the H8 weld should be six years provided that the visual examination of the top side of the H8 weld confirmed no surface flaws in the region previously inspected by UT and found to be indication free.

The above inspection has been completed and the results of the Enhanced Visual Test (EVT) have determined that the analysis assumptions associated with the top side of the H8 weld are correct with one exception. An additional indication was noted in the support ring in the HAZ of the H8 weld. This indication is approximately one foot in length in an area covered by the 1995 UT scan. This indication would have been identified by the 1995 UT scan if the indication was deeper than 0.5". Based on the 1995 UT scan and the additional visual completed, the structural integrity of the H8 weld is assured for at least one operating cycle based on the analysis submitted in the February 7, 1997 submittal for minimum ligament. The detailed analysis of the inspection results to define the required re-inspection scope and interval for the H8 weld is under study at this time. The re-inspection scope and interval for the H8 weld will be defined consistent with the BWRVIP-07 and the BWRVIP shroud support inspection and evaluation document which is under development.

During RFO14, the H8 weld was re-inspected using the OD tracker as part of the overall shroud baseline inspection performed. The same inspection coverage achieved in 1995 could not be achieved this outage because of the tie rod interferences currently present. All accessible areas using the existing H8 inspection tooling were re-inspected and a coverage of 30% was achieved. This included re-inspection of the previously identified indication on the H8 weld. This indication was located at 127 degrees and was approximately 2 inches long and 0.5" in depth on the alloy 600 cone HAZ of the H8 alloy 182 weld.

The re-inspection of 30% of the circumference confirmed no new indications and also confirmed that the previously identified indication by UT showed no growth in length or depth. The re-inspection scope achieved was therefore 66% of the baseline inspection (the 1995 baseline UT coverage was approximately 45%). This coverage is significant and is adequate to confirm that the material condition of the H8 weld, which was baselined in 1995, has not changed. The re-inspection of the H8 weld using UT was not required by the inspection plan and was considered by NMPC to be supplemental. The combined results of the required visual examination, the 1995 UT examination and the sample UT re-inspection, demonstrate that the structural integrity of the H8 weld is assured for at least one operating cycle.

The examination of weld H9 was performed as part of the inspection of components required for anchorage of the shroud repair tie rod assemblies. The exam was not intended to assess the overall structural integrity of the H9 weld. Inspection and evaluation guidelines for the H9 weld are being separately addressed by NMPC and the BWRVIP. An enhanced visual examination of the top surface of the H9 weld adjacent to one of the tie rod assemblies was performed. The examination found no indications. The same general area was inspected prior to installation of the shroud repair hardware and no indications were identified. NMPC's inspection plans for the tie rod assembly connections which included the H9 weld, was submitted to the staff in our letter of October 4, 1995 (NMP1L 0987). The staff subsequently approved the tie rod inspection plans in a safety evaluation dated March 3, 1997.

*Request for Information #5*

*In Appendix C of Inspection Summary, the description of cracking at vertical weld V9 is not consistent for the UT and VT examinations. The results of VT examinations reported that the majority of cracking is at the left hand side of heat affected zone (HAZ) on the outside diameter (OD) surface while the results of UT examinations indicate it was on the right hand side HAZ. Please clarify the inconsistency in the cracking description.*

**Required Response #5**

The UT data packages show that the majority of the V9 cracking is on the left hand of the HAZ when viewed from a perspective looking at the OD surface.

***Request for Information #6***

*Leakage Estimate*

*It appears that potential leakage from the uninspected areas assumed to be throughwall cracked were not included in the leakage estimate. If this additional leakage was included, would it affect the conclusion presented.*

- *Please provide bases for assuming a crack opening of 0.001 inch in the circumferential welds.*

- *In vertical welds, the leakage flow area was assumed to be 3 $in^2$. Please provide the bases for the assumed flow area.*

- *Was NRC approved methodology used in the leakage estimate? Please identify any assumptions made in leakage estimate that were not previously approved by NRC.*

**Required Response #6**

The leakage evaluation was performed for the welds V9 and V10 assuming throughwall cracking the entire length. The uninspected areas of welds V9 and V10 were assumed throughwall cracked.

Even if there was some additional leakage, due to postulated throughwall cracking at other vertical welds, the overall safety conclusions would remain unchanged.

- The 0.001 inch opening was a conservative representation of an IGSCC crack under axial compression. Review of metallography of cracks in boat samples taken from shroud welds at other BWRs with cracking showed that at worst the crack opening was on the order of ½ mil.

- The crack opening area was determined using fracture mechanics solutions for a through crack in a cylindrical shell by Tada and Paris described in the handbook (Reference 6) and in NUREG-3464 by Paris. The leakage flow area equaled the length of both the V9 and V10 welds times the crack opening area, with appropriate conservatism applied.

- The methodology used for this calculation is the same methodology that was used to estimate the leakage flow for the earlier NMP1 shroud repair, which the NRC

approved. An energy equation across the crack was used for the calculation with the standard values of the entrance loss coefficient, the exit loss coefficient, and the friction factor. The friction factor was calculated using Moody diagram and considered the effects of Reynolds number and the surface roughness.

*Request for Information #7*

*The LEFM analysis to determine the minimum required ligament for a vertical weld with the circumferential welds assumed failed appears to have been done using the loads from the pressure differential across the shroud and the loads from faulted conditions such as LOCA and MSLB. Much discussion is presented regarding the stress and related stress intensities from fit-up and welding. Explain why these were not included in the analysis and address their significance.*

Required Response #7

Residual stresses were not included in the LEFM assessment for the following reasons:

- For the limiting case of through wall cracking the residual stress has no impact (average stress through the thickness is zero).

- For the deeper part-through cracks the stress intensity factor due to the residual stress is compressive. Therefore ignoring this component is conservative.

- Test data from samples (from shrouds with higher fluence) show substantial ductility so that residual stresses would not contribute significantly to fracture.

Residual stresses are therefore judged to have a negligible impact on the fracture assessment and no impact on the limit load evaluation.

References:

"BWR Core Shroud Inspection and Flaw Evaluation Guidelines," GENE-523-113-0894, Rev. 2, October 1996.

"Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (BWRVIP-14)," EPRI TR-105873, March 1996. (Figure 5-3)

*Request for Information #8*

*What is the basis for the stress intensity of 20 that is used?*

Required Response #8

The K value of 20 ksi √in was the highest K calculated for the contribution from residual stresses in the Appendix B to Enclosure 1 of GENE-523-B13-01869-043, Rev. 0. As described on page B-3 and B-4, the stress intensity from figure B-6 was used in conjunction with the figure B-7 stress relaxation to determine the maximum stress intensity. The figure B-6 is a generic solution for a double v groove weld. The BWRVIP-14 has performed additional studies and has determined that this solution is representative of the core shroud welds. The assumed residual stress in the GE Appendix B evaluation is realistic and conservative and considered the effect of irradiation on the residual stresses. In addition, the vertical weld specific finite element modeling of the K value through thickness for welding plus operating plus potential fabrication related stress documented in MPM-497439 shows maximum stresses consistent with the Enclosure 1 Appendix B value of 20 ksi.

Reference:

"Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (BWRVIP-14)," EPRI TR-105873, March 1996. (Figure 5-3)

*Request for Information #9*

*When will the metallography be available for the latch that is suspected of failing by SCC? In our view the conclusion that the failure is based on SCC is speculative without the supporting information.*

Required Response #9

The damaged tie rod latches have been received by the testing facility. Preliminary results should be available by May 20, 1997. The conclusion that the failure was by SCC was based on visual observation of the fracture surface of the broken latch and comparing it with the typical fractures observed in IGSCC tests.

*Request for Information #10*

*Has an evaluation been performed that demonstrates that a sufficient ligament exists in the as found vertical welds to assure that the rings postulated in designing the shrouds would remain as rings rather than ring segments?*

Required Response #10

The ring segment between the H2 and H3 welds has been inspected by EVT for 190° of the outside surface. While the inspection could not locate the ring segment welds, V5 and V6, there was no evidence of cracking either. Furthermore, evaluation of the radial ring welds has

shown that even with relatively large cracks, the stiffness of the shroud assembly and therefore the thermal preload does not change significantly. Also, since the loading on the ring in the hoop direction is small, there are no structural integrity concerns even with significant radial weld cracking. Thus, there is high crack tolerance for the radial ring weld cracking.

*Request for Information #11*

*Has an analysis of the cracked shroud been performed assuming the tie rods were not functional (not there for analysis purposes) considering the as found condition of the shroud weld cracking? Based on such an analysis would the shroud have maintained the required ASME safety factors and therefore been considered operable for the past fuel cycle without consideration of the condition of the tie rods?*

Required Response #11

Analysis of the "as found" condition of the shroud for the H4, H5, H6a, H6b and H7 welds shows that the required safety margins would have been and continue to be maintained even if it is assumed for analysis purposes that the tie rods were not functional.

*Request for Information #12*

*Much discussion is provided regarding the fabrication factors that could have effected the residual stress distribution in the shroud that resulted in cracking taking place at the OD rather than the ID. Analytic work was reported to have been done to demonstrate this. Please provide the analysis.*

Required Response #12

The report entitled, "Analysis of Nine Mile Point Unit 1 Shroud Weld V9 and Weld V10 Cracking," by MPM Technologies was provided as an enclosure to NMPC letter NMP1L 1203 dated April 11, 1997.

*Request for Information #13*

*Information in the submittal states that hardness of the stainless shroud was limited to Rockwell "B" value of 90. Are records available of the areas that were ground? Is a reference or any other information available relating the hardness value to martensite content or to the possible reduction in the IGSCC cracking threshold?*

## Required Response #13

The hardness limitation of the plate material ($R_b$ 90) is a plate requirement in the original specification. It is included to limit the degree of cold work left in the plate after solution heat treatment. This typically ensures that the plate reached the specified temperature range for solution heat treating prior to quenching the plate. Certified Material Test Reports (CMTRs) indicated the level to average about $R_b$ 87 and thus the plates met the specification. The records will not indicate the areas ground; however, pictures of the delivered shroud, views of the enhanced visual examination in the vessel, and common practice would all indicate that much of the plate surface was ground. This is commonly found. Grinding of this type is limited to surface effects and typically influence crack initiation rather than crack growth. The impact on crack initiation is due to laps and tears on the surface, inclusions tearing out such as MnS and oxides, and sometimes phase transformations such as martensite in the areas of heaviest cold work. For example, it is very difficult to support crack initiation on a polished surface even if the material is highly sensitized. This fact is evidenced by the control of surface condition on corrosion test samples in order to obtain results that can be compared with one another.

Studies conducted at the EPRI NDE Center regarding the effects of manual surface grinding clearly showed that the depths of surface cold work were limited to about 6 or 7 mils. This is due to the fact that one cannot physically push much harder on a hand held device. At Dresden, the cold work on the core shroud ring at the core plate measured cold work effects approaching 60 mils. The reason was that large hydraulic tooling was applied to the machining process. This type of machining was not used on vertical welds at NMP1, and the depths related to hand grinding are more appropriate.

Martensite can be generated in austenitic stainless steel from grinding, and this will create local areas that will be more subject to corrosion for a number of reasons. These areas become good candidates for crack initiation. A published paper (co-authored by Dr. R. Smith, Dr. J. Danko and D. Gandy) was presented at the August 1991 NACE conference in Monterey, California entitled "Effects of Surface Preparation on Crack Initiation in Welded Austenitic Stainless Steels". The content and photos of the paper clearly show effects of all of the items mentioned above.

With regard to the use of the Rockwell hardness test to measure the cold work from grinding, it is unlikely that this test would be sufficiently sensitive to measure a hardness elevation on a 5 to 7 mil surface effect. A microhardness indentation is needed to see these effects and subsequently convert to the Rockwell scale. A value of $R_b$ 90 is too soft to indicate the presence of martensite. At Dresden the martensite, which was detected by the presence of magnetism, indicated a surface hardness of approximately $R_c$ 40, which is in the range of stellite hardfacing. Therefore the implication of the presence of martensite by the measurement of $R_b$ 90 is not valid.

Regarding an IGSCC threshold reduction as stated above, the presence of martensite will likely increase the probability of an active corrosion site but not necessarily influence a stress

threshold. No known studies are available that decipher the reason; however, the fact that corrosion is enhanced by the presence of cold work is well documented.

# ENCLOSURE 3

# REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REDESIGN OF THE LOWER WEDGE SUPPORT OF THE CORE SHROUD TIE ROD ASSEMBLY

## NON-PROPRIETARY VERSION

(Reference Letter, Darl S. Hood (NRC) to Niagara Mohawk Power Corporation, Dated April 18, 1997)

## ENCLOSURE 3

## REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REDESIGN OF THE LOWER WEDGE SUPPORT OF THE CORE SHROUD TIE ROD ASSEMBLY NINE MILE POINT NUCLEAR STATION UNIT 1

## NON-PROPRIETARY VERSION

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*Request for Information #1*

*The deflection of the C-spring in the tie rod assembly is likely to stress the latch attached to the lower wedge if it is not sliding freely. What is the estimated contribution from the C-spring deflection towards the latch stresses during heat up?*

Required Response #1

As identified in Section 3.1.2 of GENE report B13-01739-22, the vertical deflection of the C-spring from ambient conditions to full operating temperatures is . The maximum contribution to stresses in the new latch from the C-spring (assuming no sliding at the vessel wall and no loss of contact between the vessel wall and the lower wedge) will be .

*Request for Information #2*

*Provide the structural details of the nut locking device on the upper part of the tie rod assembly.*

Required Response #2

Figures 2a through 2c show details of the nut locking device. A spring retainer (Figure 2b, View A and Figure 2c, Part 8) located on the outside of the upper support bracket, when in its engaged position enables the retainer (Figure 2c, Part 6) to extend through to the inside of the upper support bracket. The retainer engages a slot in the tie rod nut as shown in Figure 2b, View F. The tie rod nut cannot rotate unless a tool is inserted to disengage the spring retainer (Figure 2b, View A) from outside the upper support bracket.

*Request for Information #3*

*Provide a clear photograph of the failure surface of the failed latch on the lower wedge to assess the failure mechanism. (The xeroxed copy provided as Figure 2 (page 26) is not sufficiently legible).*

**Required Response #3**

A photograph was provided to Mr. R. Herman of the NRC staff on April 13, 1997.

***Request for Information #4***

*You state that metallurgical examinations related to the failed latch assembly on the lower wedge will be performed to confirm that stress corrosion is the failure mechanism. When will these examinations be completed?*

**Required Response #4**

The damaged tie rod latches have been received at the testing facility. Preliminary results should be available by May 20, 1997. The conclusion that the failure was by SCC was based on visual observation of the fracture surface of the broken latch and comparing it with the typical fractures observed in IGSCC tests

***Request for Information #5***

*Explain why two latches (Alloy X-750) are used in the upper support assembly while only one is used in lower wedges. Provide details as to how they are attached to the lower springs and the lower wedges.*

**Required Response #5**

The number of latches at these locations is simply a function of geometry of the spring hardware being used at these locations. The upper spring support has two contact points and utilizes a pin geometry at the connection to the upper tie rod support hardware which necessitates having two latches to prevent either end of the pin from lifting out of the support. In contrast, the lower spring wedge has only one contact point and the geometry is such that only one latch is necessary to support the wedge. In summary, the number of latches was not determined based on loads, but rather on the amount of support needed to restrain the components from moving out of position during operation.

***Request for Information #6***

*The maximum tie rod looseness that could have been caused by the oversized holes in the shroud support cones has been calculated. Is this sufficient to cause loss of both initial and thermal preload?*

## Required Response #6

The maximum upward movement as measured by turning the tie rod nut on the 90 degree tie rod was 0.151"; whereas, the differential thermal expansion in the tie rod is 0.155" 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 wherein the measured upward movement ranged from 0.054" to 0.093".

## *Request for Information #7*

*Identify any plant operating, transient, or test conditions during which the toggle bolts on the lower tie rod anchors could slide down the oversized holes in the shroud support cones and remain there due to high frictional forces likely to exist at the hole surfaces.*

## Required Response #7

There are no plant operating, transient, or test conditions which will cause the toggle latches to slide down the shroud support cone. For this to happen, the temperature of the shroud/vessel annulus would have to be higher than the vessel and shroud temperatures, and additionally the tie rod would have to have no preload. This combination of events does not happen during operation.

## *Request for Information #8*

*It is not clear how the angle on the lower wedge is designed to increase the possibility of sliding in both directions.*

## Required Response #8

The only purpose of the angle on the lower wedge is to facilitate moving the lower wedge into position and creating the contact force with the vessel wall. The intent of the discussion in Section 3.2.1 of GENE report B13-01739-22, was to explain that the angle on the lower wedge was sufficiently small that it was not significant in influencing whether sliding occurs either at the lower wedge/lower spring interface or the vessel wall/lower wedge interface. Therefore, during plant operation, sliding is expected to occur at the surface which has the lowest friction factor. As discussed in GENE report B13-01739-22, sliding is more likely at the lower wedge/lower spring interface because the machined surfaces and dissimilar materials will result in a low friction factor.

*Request for Information #9*

*Have you considered the possibility of lowering the radial contact force at the lower wedges to assure sliding at the vessel/lower wedge and lower wedge/lower spring interfaces? What are the potential problems if the radial contact force is decreased at the lower wedge?*

Required Response #9

The intent of the lower spring design was to assure positive contact exists during shutdown conditions, and the 0.010" radial contact at the lower spring represents a minimal condition which 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" interference represents a value which confidently assures 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 degree tie rod location during the last operating cycle (which had loss of contact), and by analysis, vibration of the tie rod assembly is not a technical issue.

*Request for Information #10*

*What is the difference between condition 2 (All horizontal welds have through-wall cracks) and condition 3 (All horizontal weld cracks are 360 degrees, with no ligament remaining) identified in Section 4.0, "Consequences to Previous Plant Operation," Shroud Repair Anomalies, Nine Mile Point Unit 1, RFO14, GE Report GENE B13-01733-40?*

Required Response #10

Condition 2 identifies that "through wall" cracking occurs (depth); whereas, condition 3 addresses the circumferential length of the through wall cracking. Therefore, the combination of the two conditions provides the depth and length definition for a postulated (for analytical purposes) "360 degree through wall" crack.

*Request for Information #11*

*Provide an evaluation of the safe operation of the shroud and the tie rod assembly with the existing ligaments in the horizontal and vertical welds.*

Required Response #11

GE report GENE B13-01722-40 (Enclosure 2 of our April 8, 1997 submittal), Section 4.0, contained an evaluation that concluded the safe operation of the plant was not impaired with the as-found condition of the tie rods and an assumption that all horizontal welds were 360 degree through wall cracked. It is known based on actual inspection of the shroud that, in the regions where inspection has been performed, no through wall cracks or 360 degree crack

lengths had been found in any of the horizontal welds. Therefore, the Section 4.0 evaluation was conservative. Based on inspections of the horizontal welds completed during RFO13 and RFO14, an analysis and qualitative assessment was completed, and it was concluded that sufficient weld ligament remained such that the required safety margins would have been maintained even if it is assumed for analysis purposes that the tie rods were not functional. Additionally, Enclosure 1 of the above referenced submittal, concluded that sufficient vertical weld ligament remains such that the required safety margins would have been maintained for the shroud vertical weld as-found cracking.

*__Request for Information #12__*

*Justify the statement in the shroud repair anomalies report that the design of the latch will accommodate all potential vertical displacements without exceeding the ASME Code limits. Justify any apparent discrepancy between this and the new latch design report.*

__Required Response #12__

Table 4-1 of GENE report B13-01739-22 provides the calculated stresses for several potential sliding conditions for the new latch design which demonstrates that the stresses are less than the ASME allowable stress.

*__Request for Information #13__*

*The maximum possible differential vertical displacement of the lower wedges and the probable wedge movement have 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 should be based upon the maximum estimated vertical displacement.*

__Required Response #13__

As shown in Table 4-1 of GENE report B13-01739-22, the maximum displacement assumed for the latch design exceeds the maximum vertical displacement of the tie rod hardware at the lower spring elevation.

*Request for Information #14*

*Provide justification of structural integrity of the latches after a loss of feedwater heating event.*

**Required Response #14**

Table 4-1 of GENE report B13-01739-22 shows that the stress in the new latch for the loss of feedwater heating event are less than the ASME Code allowable.

*Request for Information #15*

*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 vessel surface and remain stuck in that position. This could impose an additional torsional moment on the latch during subsequent plant operation, and appears not to have been factored into the present design. Please discuss this potential and any considerations given to it in the design.*

**Required Response #15**

The ratio of the width of the wedge to the potential gap between the wedge and the vessel wall is . 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 of the wedge to become stuck. Even if it is assumed that the wedge were to become stuck on one edge, the latch mechanism has lateral clearance with the wedge and the spring interface slots such that the latch will reposition itself without any torsional loads being applied.